LIQUID METAL FAST BREEDER REACTORS

A Bibliography

ABOUT THE TECHNICAL INFORMATION CENTER

The Technical Information Center, as the national information center for the Department of Energy, manages, publishes, and disseminates scientific, technical, and practical energy information acquired from worldwide sources. It builds and maintains energy information data bases used to prepare abstracting journals, bibliographies, and literature searches. The data bases are also searchable via the Department's on-line retrieval system, DOE/RECON. This information is provided to DOE program offices; contractors; scientists; engineers; educational, commercial, and industrial communities; and the general public. The Technical Information Center conducts an active announcement and marketing program. It establishes policy and administers control of a DOE-publications inventory and registry. TIC also develops and evaluates new and effective information systems and services as required by scientific and technical management within the Department of Energy.

UNITED STATES DEPARTMENT OF ENERGY

Charles W. Duncan, Jr.

Secretary

Ruth M. Davis

Assistant Secretary
Resource Applications

Joseph G. Coyne

Manager
Technical Information Center

Henry D. Raleigh

Editor

About This Bibliography

The Liquid Metal Fast Breeder Reactor (LMFBR) has been under development for more than 30 years as a potential source of electric power production. Since breeder reactors utilize uranium 60 to 70 times more efficiently than light water reactors, they can extend the use of available uranium resources for centuries. Breeders are viewed by many as essential to ensure that nuclear fission will remain a long-term energy source and to reduce dependence on dwindling supplies of fossil fuels. The large number of publications on LMFBR technology attests to the importance placed on its potential for commercial power production.

This bibliography includes 5465 selected citations on LMFBR development. The citations were compiled from the DOE Energy Data Base covering the period January 1978 (EDB File No. 78R1087) through August 1980 (EDB File No. 80C79142). The references are to reports from the Department of Energy and its contractors, reports from other government or private organizations, and journal articles, books, conference papers, and monographs from U.S. originators. These citations and hundreds of additional citations on this subject are available for on-line searching and retrieval from the Technical Information Center's Energy Data Base using the DOE/RECON interactive system.

Report citations are arranged alphanumerically by report number; nonreport literature citations are arranged chronologically. Corporate, Personal Author, Subject, and Report Number Indexes are provided.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

HOW TO USE THIS BIBLIOGRAPHY

ABSTRACTS

The principal elements of abstract entries for a typical research and development report and a typical technical journal article are illustrated below.

	Report number	Title	Availability
Date of publication	Author(s)	Contract number	
1187	(TID-27639)	Gamma heated subassembly for sodium boiling experiments.	Artus, S.C. (Washington Univ., Seattle (USA)). 1975. Contract EY-76-S-05-3363. 79p. Dep. NTIS, PC A05/MF A01.

Thesis.

The design of a system to boil sodium in an LMFBR is examined. This design should be regarded as a first step in a series of boiling experiments. The reactor chosen for the design of the boiling apparatus is the Experimental Breeder Reactor-II (EBR-II), located at the National Reactor Testing Station in Idaho. Criteria broadly classified as design objectives and design requirements are discussed.

	Date of publication	Author(s)	Title	Journal citation
13569		Braun, C.	Perceptions of risks and timing in breeder development decisions.	(Electr. Power Res. Inst., Palo Alto, Calif.). <i>Am. Soc. Mech. Eng., [Pap.]</i> : No. 79-JPGC-NE-4, 1-13 (1979).

This paper considers the role of the breeder from an electric utility point of view. FBRs will be required as a capacity option, in order to maintain flexibility in the choice of the future generation mix. The uncertainties in electric demand growth rate and the availability of other power technologies indicate that foreclosing the breeder option now would not represent a prudent generation planning policy. Three conclusions are evident: There exists a need now to proceed with a well focused breeder R&D program, so that this technology is available when required. The upside risk of not having the FBR option ready on time far outweighs the downside risk of accelerated rather than reduced FBR development effort. Even with active pursuit of the FBR program, there exists sufficient time to develop and improve the institutional arrangements required to safeguard this electric generation option. 20 refs.

INDEXES

Four indexes are provided for approaching the contents of this bibliography. Each index is preceded by an introduction that details the organization of the index and the principles by which it was compiled. The reader is referred to these introductions for information not found in the index examples that follow.

•Corporate Index

Technical report literature is indexed using the name of the organization or institution responsible for the issuance of the report.

The first report cited above is indexed as:

Washington Univ., Seattle (USA)

Gamma heated subassembly for sodium boiling experiments. 1187 (TID 27639)

•Author Index

Each author's name is indexed in the form appearing on the document abstracted, with the exception that given names are reduced to initials:

Arnold, C. Jr., Preliminary evaluation of organics as coolants for LMFBR spent-fuel shipping, 1663

Artus, S.C., Gamma heated subassembly for sodium boiling experiments, 1187 (TID- 27639)

Arwas, E.B., See Smalley, A J , 964

•Subject Index

The subject index consisting of entries naming specific materials, objects, and processes is arranged alphabetically. Document titles, informative phrases, or both specific to these entries are arranged alphabetically under the entries.

EBR-2 REACTOR/REACTOR CORES

Observation of a 12-Hz effect in the upper plenum of EBR-II, 1330

EBR-2 REACTOR/REACTOR EXPERIMENTAL FACILITIES

Gamma heated subassembly for sodium boiling experiments, 1187 (TID-27639)

Physics design of an EBR-II fast-transient facility, 1594

EBR-2 REACTOR/REACTOR INSTRUMENTATION

Preliminary results of the upper-plenum acoustic experiment at EBR-II, 1635

•Report Number Index

Technical report literature is also indexed using report numbers. This index includes information on where individual reports may be obtained. Patents and conference papers are indexed here as a matter of convenience.

TID-

3333-S3	1176	Dep. NTIS, \$10.75
3333-S4	1177	Dep. NTIS, \$7.50
27313(Vol.A)	1178	AT
27313(Vol.B)	1179	AT
27313(Vol.C)	1180	AT
27313(Vol.D)	1181	AT
27315	1182	Dep. NTIS, \$7.75
27435	1183	Dep. NTIS, \$4.50
27570	1184	Dep. NTIS, PC A04/MF A01
27575	1185	Dep. NTIS, PC A04/MF A01
27579	1186	Dep. NTIS, PC A09/MF A01
27639	1187	Dep. NTIS, PC A05/MF A01

CONTENTS

ABSTRACTS 1

INDEXES

Corporate INDEX 1
Author INDEX 33
Subject INDEX 109
Report Number INDEX 335

JOURNAL TITLES LIST APPENDIX 1

ABSTRACTS

- 1 (AD-A--047410) SWELLING AND ALLOY STABILITY OF POTENTIAL FAST-BREEDER-REACTOR STRUCTURAL ALLOYS IRRADIATED IN NRL EXPERIMENT H-5. INTERIM REPORT. Reed, J.R.; Smidt, F.A. Jr. (Naval Research Lab., Washington, DC (USA)). 15 Sep 1977. Contract AT(49-5)-2110. 24p. (NRL--8130). NTIS PC A02/MF A01.
Seven commercial alloys of potential interest for fast-breeder-reactor structural alloys have been irradiated in the EBR-II to a fluence of 1.6×10^{22} n/sq cm, $E > 0.1$ MeV, at 455 C (850 F) and subsequently examined and analyzed using transmission electron microscopy. Irradiated were Inconel 718, Incoloy 800, Inconel 600, Inconel 625, PE-16, Sandvik 12R72HV, and Inconel 744. The experiment was designed to examine the alloy stability during irradiation and to measure swelling resistance. Inconel 718 and PE-16 exhibited the best swelling resistance, and Sandvik 12R72HV and Incoloy 800 had the poorest resistance to swelling. Inconel 744, an experimental superplastic alloy with a duplex microstructure, exhibited one of the highest swelling rates known for the austenite phase but showed no swelling for the ferrite phase. Inconel 625, Inconel 600, and Incoloy 800 showed extensive precipitation of phases not present prior to irradiation.

- 2 (AED-Conf--77-498-087) FUEL CYCLES OF D_2O -FAST BREEDER AND HIGH TEMPERATURE REACTORS. Zech, H.J. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Abt. fuer Angewandte Systemanalyse). (International Atomic Energy Agency, Vienna (Austria)). 1977. 28p. (CONF-7709147--50). Dep. NTIS (US Sales Only), PC A03/MF A01.
From Meeting on construction and operation management of a nuclear power plant; Karlsruhe, F.R. Germany (5 Sep 1977).
Heavy water reactors (e.g. CANDU, MZFR) show marked differences in fuelling compared to the common LWR-types. So do HTGR and FBR. This lecture summarizes the main features of these differences, and discusses their consequences. The emphasis is on heavy water reactors for power production. This survey is intended to convey general background information on fuelling, to recollect material of previous courses, and to indicate new developments in this field.

- 3 (AED-Conf--78-011-000, pp 1-2) NUCLEAR INVESTIGATIONS OF SODIUM-COOLED FAST BREEDERS WITH URANIUM-PLUTONIUM OR THORIUM-URANIUM CYCLES. Wehmann, U.; Strehlen, U. 1978. (In German).
From KTG technical meeting on the recycling of fissile materials obtained in the various reactor types from the point of view of reactor design; Juelich, F.R. Germany (24 Jan 1978).
FUEL CYCLE;LMFBR TYPE REACTORS;PLUTONIUM;THORIUM;URANIUM

- 4 (AED-Conf--78-114, pp 77) PROCEDURES USED IN THE UK FOR THE CERAMOGRAPHIC EXAMINATION OF (UPU) O_2 OXIDE FUELS FOR THE DOUNREAY PROTOTYPE FAST REACTOR. James, G. (UKAEA Reactor Group, Windscale. Reactor Development Labs.). 1978.
From Discussion meeting on characterization and quality control of nuclear fuels; Karlsruhe, F.R. Germany (13 Jun 1978).
AUTORADIOGRAPHY; CERAMOGRAPHY; COMPARATIVE EVALUATIONS; CRACKS; DATA ACQUISITION SYSTEMS; DATA COMPILATION; DISTRIBUTION; ELECTRON MICROPROBE ANALYSIS; IMAGE SCANNERS; MICROSCOPY; MICROSTRUCTURE; MIXTURES; PFR REACTOR; PHOTOMICROGRAPHY; PLUTONIUM; PLUTONIUM DIOXIDE; POROSITY; STANDARDIZATION; UNITED KINGDOM; URANIUM DIOXIDE

- 5 (AED-Conf--78-274-000, pp vp) ACTIVATION ANALYSIS FOR THE DETERMINATION OF THE SPECIFIC ACTIVITY OF RADIONUCLIDES IN METALLIC SODIUM. Stamm, H.H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Radiochemie). 1978. (In German).
From Lecture meeting on principles and applications of nuclear-, radio-, and radiation chemistry; Lindau, F.R. Germany (16 Oct 1978).
ACTIVATION ANALYSIS; IMPURITIES; LIQUID METALS; LMFBR TYPE REACTORS; MASS TRANSFER; PRIMARY COOLANT CIRCUITS; SILVER; SODIUM; ZINC

- 6 (AEEW-M--1471) WATER MODELLING STUDIES OF THE HYDRODYNAMIC CHARACTERISTICS OF IMPERMEABLE BLOCKAGES IN AN 11-PIN GEOMETRY. Robinson, D.P.; Smith, R.A.K. (UKAEA Reactor Group, Winfrith. Atomic Energy Establishment). Mar 1977. 49p. Dep. NTIS (US Sales Only), PC A03/MF A01.
A linear array of 11 pins, representing a radial slice through a 325 pin bundle, has been used to investigate the flow distribution caused by an impermeable local blockage in water. The wake geometry, which was visualised by the injection of air, is shown to be controlled by the channel flowrate and the presence of grids downstream of the blockage. From a comparison between the 11 pin results and previous investigations, the length of the wake, non-dimensionalised by a suitable blockage length scale, is shown to be consistent with measurements made with edge blockages in gridded geometries. SABRE 1 predictions, in which the 11-pin geometry is represented by a 23 subchannel model, are in excellent agreement with the measured wake lengths.

- 7 (AEEW-M--1484) WATER MODELLING STUDIES OF BLOCKAGE WITH DISCRETE PERMEABILITIES IN AN 11 PIN GEOMETRY. Robinson, D.P. (UKAEA Reactor Group, Winfrith. Atomic Energy Establishment). Jun 1977. 43p. Dep. NTIS (US Sales Only), PC A03/MF A01.

A linear array of 11 pins, representing a radial section through a 325 pin bundle, has been used to investigate the effect of discrete permeabilities on the wake geometry behind a local blockage in water. Three series of experiments were performed in each of which a different position of the permeability was considered. The complex wake geometries, visualised by the injection of air, are shown to be controlled by the position of, and flowrate through the permeability. Good agreement is shown between the experimental flow patterns and predictions by SABRE 1.

- 8 (AEEW-R--1140) TRANSMISSION OF SOUND WAVES THROUGH WATER IN A PFR SUBASSEMBLY WRAPPER. Antonopoulos Domis, M.; Butterfield, M.H. (UKAEA Atomic Energy Establishment, Winfrith). Mar 1978. 18p. Dep. NTIS (US Sales Only), PC A02/MF A01.

A theoretical model is developed which analyses the reduction of the velocity of longitudinal waves in flexible tubes of arbitrary cross section. The model is in good agreement with an experiment involving a PFR subassembly wrapper immersed in a water tank.

- 9 (AEEW-R--1168) ACOUSTIC RESONANCES AS A MEANS OF BLOCKAGE DETECTION IN SODIUM-COOLED FAST REACTORS. Domis, M.A. (Brussels Univ. (Belgium)). Mar 1978. 50p. Dep. NTIS (US Sales Only), PC A03/MF A01.

A system is described which gives early warning of the presence and build up of a local blockage in sodium cooled fast reactor sub-assemblies. The method uses acoustic resonances and reports on the analysis of signals recorded during special experiments on the Dounreay Fast Reactor. These tests proved the detectability of the standing waves and showed theoretical predictions of their frequency and shift in the presence of a blockage to be in good agreement with measurements.

- 10 (AEEW-R--1183) ZERO-POWER BAISS OF FAST REACTOR DOSIMETRY. Sanders, J.E. (UKAEA Atomic Energy Establishment, Winfrith). Jun 1978. 18p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Predictions of reaction rates, atomic displacements, and gamma-ray energy deposition in the prototype Fast Reactor are based on cross-section data and calculation methods validated against the results of zero-power experiments. The paper reviews work in Zebra relevant to this dosimetry, including neutron spectrometry, power mapping, foil activations within core heterogeneities, and measurements with thermoluminescent detectors. Comparisons of experiment and calculation are discussed in relation to the accuracies required to meet materials testing objectives.

- 11 (AERE-M--2992) BEHAVIOR OF DISSOLVED FISSION GASES PRIOR TO TRANSIENT TESTING OF FUEL PINS. Wood, M.H.; Matthews, J.R. (UKAEA Atomic Energy Research Establishment, Harwell). Oct 1978. 16p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States--UKAEA Fast Reactor Exchange Program.

The TREAT and CABRI series of reactor safety experiments on irradiated fuel require the

transfer of fuel pins from the reactor in which the fuel has achieved some burn-up to the test facility. Subsequently, the fuel is restored to power in the test facility for some time before transient heating is initiated. Such pre-test maneuvers, where the fuel is subjected to changes in the fission rate and temperature, may have important consequences for the fission gas behavior during the transient experiment. The results of rate theory calculations are used to assess these effects.

- 12 (AERE-R--8020) CONTINUOUS METHOD OF DETERMINING THE THERMODYNAMIC ACTIVITY OF CARBON IN SODIUM. Asher, R.C.; Kirstein, T.B.A.; Tolchard, A.C. (UKAEA Research Group, Harwell. Atomic Energy Research Establishment). Jan 1977. 12p. Dep. NTIS (US Sales Only), PC A02/MF A01.

The Harwell Carbon Meter (HCM) provides a continuous indication of the thermodynamic carbon activity in liquid sodium. This information is important with respect to the carburization and decarburization of steel components of the sodium circuits of sodium cooled Fast Reactors. A description of the HCM is given. It has been shown to respond rapidly to changes in carbon activity, to give an output which is in satisfactory agreement with spot calibrations using nickel 'tabs' and to operate reliably without failure for long periods.

- 13 (AERE-TP--692) BEHAVIOUR OF MATERIALS IN FAST REACTORS. Matthews, J.R. (UKAEA Atomic Energy Research Establishment, Harwell). Dec 1976. 46p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Fast neutron damage in fast reactors can limit the life of structural components through the growth of voids. The main features of the current theory of point defect production and condensation are surveyed. The role of metallurgical structures and radiation produced extended defects is outlined and used to demonstrate the development of volume swelling and radiation hardening. Mechanisms of radiation creep are described in the context of the preceding treatment of point defect behavior. Finally future trends in the field are briefly explored.

- 14 (AERE-TP--719) ANALYSIS OF CRACK PATTERNS IN FAST REACTOR FUEL PELLETS. Guha, R.M.; Head, J.L.; Matthews, J.R. (Imperial Coll. of Science and Technology, London (UK). Dept. of Mechanical Engineering; UKAEA Atomic Energy Research Establishment, Harwell). [nd]. 26p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Two dimensional (r-theta) finite element analysis is used to calculate the thermal stress in complete fast reactor fuel pellets, completely cracked pellet segments and partially cracked pellet segments. The residual stress levels are used to assess whether further cracking is probable and linear-elastic fracture mechanics used to study crack propagation. The results of the analyses show that propagation of cracks with a circumferential component becomes increasingly probable as the pellet cracks into segments with smaller angles. By making simplifying assumptions for the degree of relaxation of stress and crack healing in pellets after prolonged irradiation, the number and orientation of cooling cracks is predicted. Included are cross sections of irradiated hollow and solid fuel pellets, subjected to different rating levels.

- 15 (AERE-TP--782) MODELLING THE TRANSIENT BEHAVIOUR OF FISSION GAS. Matthews, J.R.; Wood, M.H. (UKAEA Atomic Energy Research Establishment, Harwell). Feb 1979. 31p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States-UKAEA Fast Reactor Exchange Program.

To gain an understanding of the behavior of fuel during hypothetical accident situations rapid heating experiments are in progress in pulsed reactors such as TREAT and in out-of-pile facilities. The role of the fission gases is particularly important in such accident situations and it is their behavior which is investigated. The behavior of intragranular gas during transient situations is discussed. A simple model is constructed of this which accounts for bubble movement in a temperature gradient, bubble growth through the capture of single gas atoms and coalescence among the bubble population, and the non-equilibrium character of the bubbles. The release of gas to the grain boundaries and intragranular swelling is determined. The model has been incorporated into the fuel behavior code, FRUMP, and has been applied both to out-of-pile heating experiments and the in-pile TREAT test H3.

- 16 (AERE-TP--793) COMPARISON OF EXPERIMENT AND NEFIG MODEL CALCULATIONS OF TRANSIENT FISSION GAS BEHAVIOR. Wood, M.H.; Matthews, J.R.; Matthews, H.R. (UKAEA Atomic Energy Research Establishment, Harwell). Jun 1979. 29p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States-UKAEA Fast Reactor Exchange Program.

Calculations have been performed, using the NEFIG model of transient gas behavior, of rapid external heating experiments performed on pre-irradiated mixed oxide fuel. The calculations predict fission gas release, intragranular bubble sizes and swelling in good agreement with experiment.

- 17 (AERE-TP--796) FAST REACTOR FUEL PIN BEHAVIOUR MODELLING IN THE UK. Matthews, J.R.; Hughes, H. (UKAEA Atomic Energy Research Establishment, Harwell). May 1979. 23p. Dep. NTIS, PC A02/MF A01.

Work performed under United States-UKAEA Fast Reactor Exchange Program.

Two fuel behavior codes have been applied extensively to fast reactor problems; SLEUTH developed at Springfields Nuclear Laboratory and FRUMP at AERE Harwell. Other UKAEA Establishments and those of the CEGB have contributed work which has been important in model development. The codes themselves are available for use by the various organizations concerned with fast reactors.

- 18 (AI-DOE--13208) SODIUM TECHNOLOGY AND COVER GAS SEAL DEVELOPMENT PROGRAMS. QUARTERLY TECHNICAL PROGRESS REPORT, JULY--SEPTEMBER 1977. (Atoms International Div., Canoga Park, CA (USA)). 10 Nov 1977. Contract EY-76-C-03-0824. 64p. AT.

Progress is summarized in studies related to LMFBR-FFTF cover gas seals development and testing and in sodium removal and decontamination of LMFBR components. (DG)

- 19 (AI-DOE--13209) FAST REACTOR FLUENCE DOSIMETRY. TECHNICAL PROGRESS REPORT, JUNE--DECEMBER 1977. (Atoms International Div., Canoga Park, CA (USA)). 31 Mar 1978. Contract EY-76-C-03-0824-027. 30p. Dep. NTIS, PC A03/MF A01.

The objectives of this task are to develop and demonstrate the use of ^{10}B and ^6Li helium accumulation fluence monitors (HAFM's) as a reliable and accurate method of measuring reactor neutron fluence, develop and apply an expanded set of HAFM's that will provide fluence responses in different but overlapping neutron energy ranges, and identify, through the precise measurement of spectrum-integrated helium production cross sections, those elements producing significant helium when used individually or as components of advanced alloys in FTR and breeder reactor neutron environments, so that their use might be eliminated, minimized, or controlled.

- 20 (AI-DOE--13210) LMFBR SAFETY PROGRAM. ANNUAL TECHNICAL PROGRESS REPORT. GOVERNMENT FISCAL YEAR, 1977. (Atoms International Div., Canoga Park, CA (USA)). 15 Dec 1977. Contract EY-76-C-03-0824-010. 40p. Dep. NTIS, PC A03/MF A01.

Information is presented concerning the development of the SOMIX-1 computer code for sodium drop burning analysis; experimental analysis of burning sodium drops; aerosol leakage from containment buildings; high-temperature-concentration aerosols; aerosol source term from vaporized fuel; properties of high-temperature fuel mixtures; and development of the COMRADEX computer code for analysis of radiological doses in the environment from LMFBR accidents.

- 21 (AI-DOE--13211) SODIUM-TO-GAS LEAK DETECTION. ANNUAL TECHNICAL PROGRESS REPORT, GOVERNMENT FISCAL YEAR 1977. (Atoms International Div., Canoga Park, CA (USA)). 18 Nov 1977. Contract EY-76-C-03-0824-012. 12p. AT.

The objectives of the program are to: (1) demonstrate the sensitivity, reliability and operability of liquid metal-to-gas leak detection instrumentation, and (2) provide information that can be used to establish criteria to guide the design of detection systems for use in LMFBR plants. During the report period, this project continued to provide data that are of use to the Clinch River Breeder Reactor Plant (CRBRP), Fast flux Test Facility (FFTF) and the American Society of Mechanical Engineers (ASME) working group on inservice inspection. These data were derived from: (1) small insulated pipe (test chamber atmosphere natural circulation) sodium leak tests, and (2) long-term performance tests of aerosol detectors at EBR-II. Preparations were made for tests to verify the performance of selected aerosol detectors in simulated LMFBR environments. Testing was postponed in deference to urgent experimental efforts essential to FFTF leak detection system design.

- 22 (AI-DOE--13212) LMFBR FLEXIBLE PIPE JOINT DEVELOPMENT PROGRAM. ANNUAL TECHNICAL PROGRESS REPORT, GOVERNMENT FISCAL YEAR 1977. (Atoms International Div., Canoga Park, CA (USA)). 15 Feb 1978. Contract EY-76-C-03-0824-019. 131p. Dep. NTIS, PC A07/MF A01.

Currently, the ASME Boiler and Pressure Vessel Code does not allow the use of flexible pipe joints (bellows) in Section III, Class 1 reactor primary piping systems. Studies have shown that the primary piping loops of LMFBR's could be simplified by using these joints. This simplification translates directly into shorter primary piping runs and reduced costs for the primary piping system. Further cost savings result through reduced vault sizes and reduced containment building diameter. In addition, the use of flexible joints localizes the motions

from thermally-induced piping growth into components which are specifically designed to accommodate this motion. This reduces the stress levels in the piping system and its components. It is thus economically and structurally important that flexible piping joints be available to the LMFBR designer. The overall objective of the Flexible Joint Program is to provide this availability. This will be accomplished through the development of ASME rules which allow the appropriate use of such joints in Section III, Class 1 piping systems and through the development and demonstration of construction methods which satisfy these rules. The rule development includes analytic and testing methodology formulations which will be supported by subscale bellows testing. The construction development and demonstration encompass the design, fabrication, and in-sodium testing of prototypical LMFBR plant-size flexible pipe joints which meet all ASME rule requirements. The satisfactory completion of these developmental goals will result in an approved flexible pipe joint design for the LMFBR. Progress is summarized in the following efforts undertaken during 1977 to accomplish these goals: (1) code case support, (2) engineering and design, (3) material development, (4) testing, and (5) manufacturing development.

- 23 (AI-DOE--13214) HIGH-TEMPERATURE PIPING DESIGN TECHNOLOGY. QUARTERLY TECHNICAL PROGRESS REPORT, JULY--SEPTEMBER 1977. Jetter, R.L.; Naugle, F.V. (Atomics International Div., Canoga Park, CA (USA)). 14 Dec 1977. Contract EY-76-C-03-0824-021. 49p. Dep. NTIS, PC A03/MF A01.

Progress is reported in the following LMFBR piping design studies: collection of sodium piping failure data, analysis of piping systems with prior successful operating history, evaluation of sensitivity of piping design margins to variability in materials and geometry, and high-cycle fatigue behavior at elevated temperature.

- 24 (AI-DOE--13216) EVALUATIONS OF THE PIPING SYSTEM INELASTIC ANALYSIS COMPUTER PROGRAM PIRAX2. Rodabaugh, E.C. (Atomics International Div., Canoga Park, CA (USA)). 24 Oct 1977. Contract EY-76-C-03-0824. 75p. Dep. NTIS, PC A04/MF A01.

The report contains two sets of comparisons of inelastic test data with PIRAX2-theory; i.e., ORNL beam tests and ORNL elbow tests. The purpose of these comparisons is to evaluate the accuracy of the simplified analytical techniques used in PIRAX2. The test data are on structures that are much simpler than piping systems but provide a fundamental basis for comparison. The report includes an analysis of a 3-anchor piping system to illustrate the relative simplicity of PIRAX2 input/output data and relatively small computer running time. Some areas of needed improvements in PIRAX2 are discussed.

- 25 (AI-DOE--13218) DECONTAMINATION OF LMFBR COMPONENTS. Hill, E.F. (Atomics International Div., Canoga Park, CA (USA)). 31 Jan 1978. Contract EY-76-C-03-0824. 124p. AT.

A decontamination process specification for use on Type 304 stainless steel hot leg components of an LMFBR system has been issued. It is based on the use of 2.5 percent glycolic--2.5 percent citric acids at a temperature of 70°C. The criteria used for selecting this system are described, along with the supporting metal removal and metallurgical data. The

results of decontamination tests using radioactive specimens from two different loops are presented. The reagent selected exhibits superior decontaminating capability with acceptable metal damage.

- 26 (AI-DOE--13220) QUARTERLY TECHNICAL PROGRESS REPORT. SODIUM TECHNOLOGY AND COVER GAS SEAL DEVELOPMENT PROGRAMS. OCTOBER--DECEMBER 1977. (Rockwell International Corp., Canoga Park, CA (USA). Rocketdyne Div.). 15 Feb 1978. Contract EY-76-C-03-0824-027. 21p. AT.

Progress is summarized in FFTF-LMFBR cover gas seals testing and design and in sodium removal and decontamination techniques. (DG)

- 27 (AI-DOE--13221) QUARTERLY TECHNICAL PROGRESS REPORT COVER GAS SEAL COMPONENTS AND SODIUM COMPONENT DEVELOPMENT PROGRAMS. OCTOBER--DECEMBER 1977. (Rockwell International Corp., Canoga Park, CA (USA). Rocketdyne Div.). 30 Jan 1978. Contract EY-76-C-03-0824-026. 20p. AT.

Progress is summarized in LMFBR cover gas seals development and testing and in studies of sodium wetting of fuel handling subcomponents. (DG)

- 28 (AI-DOE--13222) HIGH-TEMPERATURE PIPING DESIGN TECHNOLOGY. QUARTERLY TECHNICAL PROGRESS REPORT, OCTOBER--DECEMBER 1977. (Rockwell International Corp., Canoga Park, CA (USA). Rocketdyne Div.). 15 Feb 1978. Contract EY-76-C-03-0824-026. 11p. AT.

Progress is summarized in the following LMFBR piping design studies: collection of sodium piping failure data, analysis of piping systems with prior successful operating history, evaluation of sensitivity of piping design margins to variability in materials and geometry, and high cycle fatigue behavior at elevated temperature. (DG)

- 29 (AI-DOE--13223) REACTOR SAFETY. QUARTERLY TECHNICAL PROGRESS REPORT, OCTOBER--DECEMBER 1977. (Atomics International Div., Canoga Park, CA (USA)). 21 Mar 1978. Contract EY-76-C-03-0824-028. 27p. Dep. NTIS, PC A02/MF A01.

The objectives of the program are to conduct tests that will characterize the behavior of sodium oxide, fuel, fission product, and other aerosols as they might be generated by various postulated LMFBR accidents; determine by analysis and confirm by experiment the generation and transport of these aerosols with respect to source (location, type, and configuration), for the entire course of events associated with real and hypothetical accident conditions; and conduct tests that will determine the effect of molten fuel on reactor structural or sacrificial material.

- 30 (AI-DOE--13225) SODIUM PIPING SURVEY TO JULY 1977. Novacek, P.F. (Atomics International Div., Canoga Park, CA (USA)). 22 Feb 1978. Contract EY-76-C-03-0824. 112p. AT.

This document presents a survey of high-temperature sodium piping experience through July 1, 1977. This survey was developed from reported data and from personal contacts with people familiar with specific installations. The survey provides information on piping that had no failures as well as piping that did. Twenty-six failures were identified in > or = to 1-in. diameter piping with only one being directly related to piping design.

- 31 (AI-DOE--13226) PERFORMANCE TESTS OF SODIUM DIP SEALS. Shimazaki, T.T. (Atoms International Div., Canoga Park, CA (USA)). 28 Feb 1978. Contract EY-76-C-03-0824. 110p. AT.

Tests made on the CRBR sodium dip seal show that it meets all its capability requirements, provided the 400°F dip seal walls are wetted. It was demonstrated that the dip seal walls can be wetted at 400°F by either initially tin-plating the surface at room temperature by the brush electroplating process or by subjecting the submerged 400°F surface to ultrasonic energy. Data were also obtained on argon convection and sodium frost deposition in the upper annulus, corrosion of SA508 Class II steel in sodium, mass transfer coefficient for O₂ in sodium, and effect of solids buildup in dip seal on plug rotational binding.

- 32 (AI-DOE--13227) SELECTION OF A SODIUM-AND RADIATION-RESISTANT SEALANT FOR LMFBR EQUIPMENT CELL PENETRATIONS. Humphrey, L.H.; Horton, P.H. (Atoms International Div., Canoga Park, CA (USA)). 31 Jan 1978. Contract EY-76-C-03-0824-001. 44p. AT.
- The test program is described which selected a sealant for LMFBR equipment cell penetrations capable of withstanding 1000°F sodium for 2 h after being exposed to 10⁶ rads of gamma radiation. The selected material is Dow-Corning 93-104.

- 33 (AI-DOE--13228) CRBRP ADVANCED FUEL RELOAD CORE DESIGN STUDIES. SUMMARY REPORT. Vitti, J.A.; Galluzzo, N.G.; Brittingham, J.C.; Ostermier, B.J.; von Arx, A.V.; Moldenhauer, J. (Atoms International Div., Canoga Park, CA (USA)). 15 May 1978. Contract EY-76-C-03-0824. 288p. AT.

Trade studies were performed based on the identified constraints, ground rules, and assumptions, to establish preferred preconceptual CRBRP core designs using advanced fuel. These designs were subdivided into Level I and Level II categories. The Level I concepts primarily involved a switch of fuel and blanket subassemblies only, compared to the reference (U-Pu)O₂ homogeneous and heterogeneous cores. The Level II designs not only used advanced fuel but also involved cost- and performance-effective modifications of components not normally removed at refueling. Two Level I and three Level II designs were selected. These core designs are denoted by the acronyms RC1S, AC1N, RC2S, AC2N, and AC2S, which are explained in Table 2.1. Of these designs, the tightly-coupled heterogeneous AC2N concept is recommended as the conceptual core design for the CRBRP when fueled with (U-Pu)C. The AC2N core design offers a high breeding ratio, low doubling time, reasonably low inventory, and good economics, while maintaining a low sodium void worth. It is, therefore, an attractive alternative for use as a CRBRP reload core.

- 34 (AI-DOE--13231) SODIUM TECHNOLOGY AND COVER GAS SEAL DEVELOPMENT PROGRAMS. QUARTERLY TECHNICAL PROGRESS REPORT, JANUARY--MARCH 1978. (Atoms International Div., Canoga Park, CA (USA)). 30 Apr 1978. Contract EY-76-C-03-0824-025. 41p. AT.

Progress is reported in the following study areas: sodium evaporation process development, decontamination studies, advanced seal materials, and gamma irradiation effects on seal materials. Some of the studies were aimed specifically at FFTF or CRBR component development.

- 35 (AI-DOE--13232) COVER GAS SEAL COMPONENTS AND SODIUM COMPONENT DEVELOPMENT PROGRAMS. QUARTERLY TECHNICAL PROGRESS REPORT, JANUARY--MARCH 1978. (Atoms International Div., Canoga Park, CA (USA)). 30 Apr 1978. Contract EY-76-C-03-0824-011. 18p. AT.

Progress is reported in the following study areas: CRBRP inflatable seal vendor qualification, static inflatable seal development, high-temperature seal degradation, and sodium wetting tests on specimen materials.

- 36 (AI-DOE--13233) HIGH-TEMPERATURE PIPING DESIGN TECHNOLOGY. QUARTERLY TECHNICAL PROGRESS REPORT, JANUARY--MARCH 1978. (Atoms International Div., Canoga Park, CA (USA)). 30 May 1978. Contract EY-76-C-03-0824-026. 24p. AT.

Progress is summarized in the following LMFBR piping design studies: (1) collection of piping failure and construction defect data; (2) analysis of systems with prior successful operating history; (3) evaluation of sensitivity of piping design margins to variability in materials, loading, temperature, and geometry; (4) high cycle fatigue behavior at elevated temperature; and (5) piping component testing. (DG)

- 37 (AI-DOE--13234) REACTOR SAFETY QUARTERLY TECHNICAL PROGRESS REPORT, JANUARY--MARCH 1978. (Atoms International Div., Canoga Park, CA (USA)). 31 May 1978. Contract EY-76-C-03-0824-028. 19p. Dep. NTIS, PC A02/MF A01.

Progress is summarized in LMFBR safety studies related to accident debris behavior. Included are sections on aerosol leakage, large scale molten fuel tests, and coordination of LMFBR risk analysis and safety studies. (DG)

- 38 (AI-DOE--13236) FAST REACTOR FLUENCE DOSIMETRY TECHNICAL PROGRESS REPORT FOR PERIOD JANUARY--MARCH 1978. (Atoms International Div., Canoga Park, CA (USA)). 18 May 1978. Contract EY-76-C-03-0824-027. 38p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Progress is summarized in studies and tests related to the development and demonstration of ¹⁰B and ⁶Li helium accumulation fluence monitors as a reliable and accurate method of measuring LMFBR neutron fluence. (DG)

- 39 (AI-DOE--13238) SELECTION OF BEARING MATERIALS FOR COMMERCIAL LMFBR FLEXIBLE JOINT APPLICATIONS. Kuster, C.A.; Sarraf, R.J.; Tsang, S. (Atoms International Div., Canoga Park, CA (USA)). 7 Jul 1978. Contract EY-76-C-03-0824-019. 73p. AT.

Results of the verification and test program conducted for selecting a bearing couple for hinge assemblies of the angular flexible joint for use in primary LMFBR piping are reported. Initial screening phase friction and wear tests and thermal cycling tests were performed on 13 bearing material combinations in a nitrogen environment. These bearing combinations, selected from a wide range of bearing materials and combinations studied in a literature survey, were considered to be the most promising to meet the requirements of the system. Final friction and wear testing of the best four bearing combinations were conducted in both nitrogen and air environments. Two bearing combinations, K162B/K162B and Tribaloy 700/Colmonoy 6, are recommended for further evaluation in prototypic testing.

- 40 (AI-ERDA--13198) FAST REACTOR FLUENCE DOSIMETRY. TECHNICAL PROGRESS REPORT, DECEMBER 1976--MAY 1977. (Atomics International Div., Canoga Park, CA (USA)). 31 Aug 1977. Contract EY-76-C-03-0824-002. 26p. AT.
Progress is summarized in studies and tests related to the development and demonstration of ^{10}B and ^6Li helium accumulation fluence monitors as a reliable and accurate method of measuring LMFBR neutron fluence. (DG)

- 41 (AI-ERDA--13204) SODIUM TECHNOLOGY AND COVER GAS SEAL DEVELOPMENT PROGRAMS. QUARTERLY TECHNICAL PROGRESS REPORT, APRIL--JUNE 1977. (Atomics International Div., Canoga Park, CA (USA)). 29 Jul 1977. Contract EY-76-C-03-0824-001. 53p. AT.
Progress is summarized in LMFBR-FFTF seals testing, cover gas seal components development, and sodium removal and component decontamination techniques. (DG)

- 42 (AI-ERDA--13205) LMFBR SAFETY PROGRAM. QUARTERLY TECHNICAL PROGRESS REPORT, APRIL--JUNE 1977. Heisler, M.P.; Johnson, R.P.; Nelson, C.T.; Vaughan, E.U.; Guderjohn, C. (Atomics International Div., Canoga Park, CA (USA)). 31 Aug 1977. Contract EY-76-C-03-0824-019. 24p. Dep. NTIS, PC A02/MF A01.
Progress is summarized in LMFBR safety studies related to the characterization of sodium fires and fission products. Included are sections on sodium jet dispersal, aerosol leakage, high temperature-concentration aerosols, characteristics of released aerosols, and SOMIX code development. (DG)

- 43 (AI-ERDA--13206) HIGH-TEMPERATURE PIPING DESIGN TECHNOLOGY. QUARTERLY TECHNICAL PROGRESS REPORT, APRIL--JUNE 1977. Jetter, R.I.; Naugle, F.V. (Atomics International Div., Canoga Park, CA (USA)). 15 Aug 1977. Contract EY-76-C-03-0824-021. 23p. Dep. NTIS, PC A02/MF A01.
Progress is summarized in the following LMFBR piping design studies: long-range program planning, collection of piping failure and construction defect data, analysis of systems with prior successful operating history, and compilation of fatigue data for 2-1/4 Cr-1 Mo and type 316 stainless steels. (DG)

- 44 (AI-ERDA--13207) WATER TESTING OF THE INDUCER PUMP. Dunn, C.; Hoshide, R.K. (Atomics International Div., Canoga Park, CA (USA)). 15 Sep 1977. Contract EY-76-C-03-0824-024. 100p. Dep. NTIS, PC A05/MF A01.
The inducer pump, designed and fabricated as a test article to evaluate the inducer/impeller pump concept for providing improved suction performance of large sodium pumps, met or exceeded all performance goals. The inducer stage in front of the centrifugal impeller dramatically reduces the size of pumps. It was demonstrated that the inducer can operate at suction specific speeds in excess of 35,000, whereas conventional centrifugal pumps have demonstrated capability to only 12,000. The inducer pump employs several components of the 2000 and 4000 gpm NaK pumps from the Connecticut Aircraft Nuclear Engine Laboratory (CANEL) Programs. Use of the volute, thermal barrier and shaft oil seal from the CANEL pumps allowed acceleration and cost reduction of the Inducer Pump Program.

- 45 (AI/FBR--77-1) PLBR CORE SELECTION FOR

LOW CDA ENERGETICS: LIQUID METAL FAST BREEDER REACTOR DESIGN PROJECT. (Atomics International Div., Canoga Park, CA (USA)). 1 Feb 1977. Contract EY-76-C-03-1142. 306p. AT.

A study made to determine the feasibility of providing a PLBR core which has inherently nonenergetic CDA characteristics is presented. If such a core can be provided, it is expected that containment of large CDA energetics would not be required for licensing. The five major stages of this effort were: (1) Establishment of design requirements and configuration selection criteria; (2) Performance of extensive nuclear design and reactor system-related safety parametric studies, to provide a basis for choosing designs (i.e., bullseye and spoiled regular configurations) representative of the basic "nonenergetic" core concepts; (3) Selection and detailed characterization of the reference bullseye and spoiled regular core configurations; (4) Comparison of the reference designs against the selection criteria; and (5) Selection of the core configuration to be recommended for PLBR. Also included in the study was the determination of characteristics of a regular core. This provided a baseline against which to compare the performance and safety features of the candidate low-energy cores. Details of the approach followed, and results obtained, are presented.

- 46 (ANL--75-41) RESPONSE OF EBR-II TO REACTIVITY INSERTION. Dean, E.M.; Sackett, J.I. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 45p. AT.

The report presents the results of a parametric study to determine how variations in important reactor parameters affect the EBR-II dynamic response to reactivity change. The purpose of the study was to (1) define the inherent capabilities of the plant protection system (PPS), (2) identify essential trip parameters and their range of applicability, and (3) identify reactor parameters or conditions which, if changed, would affect the performance of the PPS. As a result of this study, the bases for reactivity-related limits in EBR-II were developed.

- 47 (ANL--76-34) FINAL SAFETY ANALYSIS ADDENDA TO HAZARDS SUMMARY REPORT, EXPERIMENTAL BREEDER REACTOR II (EBR-II): UPGRADING OF PLANT PROTECTION SYSTEM. VOLUME I. Sackett, J.I.; Gale, N.L. (comps.). (Argonne National Lab., IL (USA)). Aug 1977. Contract W-31-109-ENG-38. 174p. AT.

This report is a compilation of the formal Final Safety Analysis Addenda (FSAA's) to the EBR-II Hazard Summary Report and Addendum that have been prepared in support of certain modifications to the reactor-shutdown-system portion of the EBR-II plant protection system. Each major section is an edited version of the original FSAA for a particular modification and provides a description of the pre- and post-modification system, the rationale for the modification, and required supporting safety analysis.

- 48 (ANL--76-44) EXAMINATION AND EVALUATION OF IN-REACTOR FRACTURE OF SHROUD TUBES IN MIXED-OXIDE-FUEL EXPERIMENT X159. Flinn, J.E.; Cook, J.H.; Koenig, J.F.; Williamson, M.H. (Argonne National Lab., IL (USA)). Jun 1977. Contract W-31-109-ENG-38. 43p. AT.

During disassembly and subsequent visual examinations of X159 (a Mark A-19A type of EBR-2 subassembly containing mixed-oxide fuel), 11 of 19 shroud tubes were found fully or partially severed. Several of the capsules

within their shroud tubes were distinctly kinked at axial locations near those at which the shroud tubes were severed. The examination also disclosed that the shroud-tube fractures occurred during reactor operation. The unirradiated shroud-tube material and some of the fracture areas of the irradiated material were structurally examined and evaluated. The surface appearances of the two materials suggested that the shroud-tube failures were produced by stress corrosion. The conditions for promoting stress corrosion could have easily been present in the subassembly. Possible stress-producing mechanisms associated with the reactor environment were analyzed. These mechanisms predicted conditions that did not correlate with the observations. It was concluded that kinks in several of the fuel-element capsules were the more logical candidates for producing the shroud-tube fractures. What produced the kinks was not resolved.

- 49 (ANL--76-78) DESIGN AND FABRICATION OF THE EBR-II ENVIRONMENTAL INSTRUMENTED SUBASSEMBLY: TEST XX07. Gillette, J.L.; Poloncsik, J.; Smaardyk, A.; Walker, D.E.; Filewicz, E.C.; Longnecker, A.A.; Murphy, W.A.; Wendte, J.C. (Argonne National Lab., IL (USA)). Nov 1977. Contract W-31-109-ENG-38. 65p. AT.

The EBR-II environmental instrumented subassembly (EISA), or Test XX07, was designed, fabricated, and irradiated to provide in-core measurements of irradiation conditions. This subassembly contained 57 elements with uranium metal-5 wt % fissium fuel and four nonfueled elements. It was instrumented with two permanent-magnet flowmeters, 13 coolant thermocouples at various axial and radial locations, 10 fuel-centerline thermocouples, and two self-powered detectors. The subassembly was irradiated in EBR-II for about 11,600 MWd (approximately 186 days), reaching a peak burnup of about 2.9 at. %. During its residence in the reactor, XX07 provided in-core measurements of flow, temperature, and flux levels under both steady-state and dynamic conditions. This report discusses the detailed design of XX07 and the various techniques used in its fabrication and assembly. The many component parts of XX07, as well as the final, completed subassembly, are described.

- 50 (ANL--76-94) EFFICACY OF THE EBR-II FERD SYSTEM AS AN AUTOMATIC TRIP DEVICE. Fryer, R.M.; Dean, E.M.; Lambert, J.D.B.; Reyes, L.A. (Argonne National Lab., IL (USA)). Aug 1977. Contract W-31-109-ENG-38. 72p. AT.

The effectiveness of the EBR-II fuel-element-rupture-detector (FERD) system which detects delayed-neutron-emitting fission products is discussed. It is demonstrated that FERD provides no reactor or public protection for loss-of-flow or transient-overpower fault events that affect the whole core; such protection is provided by other systems. For single-element faults (cladding breaches), protection is not required, because the failures are nonpropagative. For intermediate fault events, for example, flow blockage to a single subassembly, the element cladding, sodium coolant, and hexagonal subassembly duct provide effective barriers to potential failure propagation. The FERD system provides beneficial diagnostic information about such events, but is ineffective in preventing either the original event or potential subsequent rapid propagative events when used as an automatic trip device.

- 51 (ANL--77-11) DATA AND ANALYSES FOR INCONEL X750 SPRINGS IRRADIATED IN SURV-1, -3, -4, -5, AND -6. Walters, L.C.; Ruther, W.E. (Argonne National Lab., IL (USA)). Feb 1978. Contract W-31-109-ENG-38. 42p. AT.

In 1965, eight surveillance subassemblies were placed in row 12 of EBR-II, where the irradiation temperature would be near the sodium-inlet temperature of 371°C. At the same time, two other surveillance subassemblies were placed in the primary-tank storage basket, which receives minimal neutron exposure but is immersed in primary sodium and experiences a temperature of 371°C. Each subassembly contained 18 preloaded springs made of Inconel X750. Springs from four of the in-core subassemblies and from one subassembly in the storage basket have been evaluated to determine rates of irradiation-enhanced deformation at neutron exposures of up to 4.2 dpa. The creep coefficient derived from the stress-relaxation measurements on the springs was 1.0×10^{-12} (Pa · dpa)⁻¹ for neutron exposures of 4.2 dpa (3751 days) at an in-reactor temperature of 371°C. The relaxation behavior was adequately described by a creep equation that was linear in neutron fluence and applied stress. Springs encapsulated in helium showed in-reactor relaxation rates identical to those of springs exposed to the flowing primary sodium. The creep coefficient derived from this work in Inconel X750 springs was almost the same as the creep coefficients determined for various austenitic stainless steel alloys.

- 52 (ANL--77-15) TH13D-1: A COMPUTER PROGRAM FOR STEADY-STATE THERMAL-HYDRAULIC MULTICHANNEL ANALYSIS. Sha, W.T.; Schmitt, R.C.; Lin, E.I.H. (Argonne National Lab., IL (USA)). Jul 1977. Contract W-31-109-ENG-38. 107p. AT.

TH13D-1 is an improved version of the TH13D computer program for steady-state, single-phase, thermal-hydraulic multichannel analysis. The program accounts for conservation of mass, energy, and momentum subject to pressure-drop boundary conditions, and leads to a nonlinear multipoint boundary-value problem. Turbulent interchange, radial thermal conduction, and forced flow due to wire wraps or grids between channels are explicitly taken into account. Temperature distributions in the coolant, cladding, fuel, and duct wall and the size of the central void of the oxide fuel after thermal restructuring are computed. Three different thermal-hydraulic channel arrangements (square, hexagonal, and triangular) can be treated. Multipin analysis with transverse interactions or multiassembly calculations without transverse interactions between the channels can be performed. In addition, (1) a new wire-wrap model has been developed that accounts for most measured mixing data in a wire-wrapped rod bundle; (2) bypass fluid of a fuel assembly is accounted for; (3) an implicit option for the energy equation is provided (this option is necessary for low Reynolds number flow); and (4) an improved transverse-momentum equation is incorporated. Also included are program-input description and format, and a sample problem reflecting these improvements.

- 53 (ANL--77-20) BUBBLE DYNAMICS IN A SUPERHEATED LIQUID. Sha, W.T.; Shah, V.L. (Argonne National Lab., IL (USA)). Sep 1977. Contract W-31-109-ENG-38. 55p. (NUREG--0333). Dep. NTIS, PC A04/MF A01.

The report presents an extensive literature survey on bubble dynamics. Growth of a single spherical bubble moving in a uniformly superheated liquid is considered. Equations of

motion and energy are presented in the forms that take into consideration the interaction between the motion and the growth. The fourth-order Runge-Kutta method is used to obtain a simultaneous solution of equations of motion and growth rate, and the solution is compared with available experimental results. Results for liquid sodium are presented for a range of pressures and Jakob numbers.

- 54 (ANL--77-25) FINAL REPORT OF FUEL DYNAMICS TEST E7. Doerner, R.C.; Murphy, W.F.; Stanford, G.S.; Froehle, P.H. (Argonne National Lab., IL (USA)). Apr 1977. Contract W-31-109-ENG-38. 150p. Dep. NTIS, PC A08/MF A01.

Test data from an in-pile failure experiment of high-power LMFBF-type fuel pins in a simulated \$3/s transient-overpower (TOP) accident are reported and analyzed. Major conclusions are that (1) a series of cladding ruptures during the 100-ms period preceding fuel release injected small bursts of fission gas into the flow stream; (2) gas release influenced subsequent cladding melting and fuel release [there were no measurable FCI's (fuel-coolant interactions), and all fuel motion observed by the hodoscope was very slow]; (3) the predominant postfailure fuel motion appears to be radial swelling that left a spongy fuel crust on the holder wall; (4) less than 4 to 6 percent of the fuel moved axially out of the original fuel zone, and most of this froze within a 10-cm region above the original top of the fuel zone to form the outlet blockage. An inlet blockage approximately 1 cm long was formed and consisted of large interconnected void regions. Both blockages began just beyond the ends of the fuel pellets.

- 55 (ANL--77-27) PHYSICS CALCULATIONS FOR THE CLINCH RIVER BREEDER REACTOR. Kalimullah, Kier, P.H.; Hummel, H.H. (Argonne National Lab., IL (USA)). Jun 1977. Contract W-31-109-ENG-38. 176p. Dep. NTIS, PC A09/MF A01.

Calculations of distributions of power and sodium void reactivity, unvoided and voided Doppler coefficients and steel and fuel worths have been performed using diffusion theory and first-order perturbation theory for the LWR discharge Pu-fueled CRBR at BOL, the FFTF-grade Pu-fueled CRBR at BOL and for the beginning and end of equilibrium cycle of the LWR-Pu-fueled CRBR. The results of the burnup and breeding ratio calculations performed for obtaining the reactor compositions during the equilibrium cycle are also reported. Effects of sodium and steel contents on the distributions of sodium void reactivity and steel worth have also been studied. Errors and uncertainties in the reactivity coefficients due to cross-sections and the two-dimensional geometric representations of the reactor used in the calculations have also been estimated. Comparisons of the results with those in the CRBR PSAR are also discussed.

- 56 (ANL--77-40) SUMMARY AND EVALUATION OF TRANSIENT TEST DATA ON EXTENDED FUEL MOTION RESULTING FROM UNPROTECTED LOSS OF FLOW. Dickerman, C.E. (Argonne National Lab., IL (USA)). Jun 1977. Contract W-31-109-ENG-38. 38p. Dep. NTIS, PC A03/MF A01.

Experimental data on fuel failure and its consequences are used to provide predictions of the extended fuel motion that would occur after a hypothetical core-disruptive loss-of-flow accident without scram in the Fast Test Reactor. These data indicate the existence of fuel-dispersal mechanisms, slow collapse of fresh fuel, resistance to collapse of

irradiated fuel, the absence of accelerated collapse mechanisms (such as cladding-coolant vapor "explosions"), and the fact that real LMFBF conditions inhibit rapid sodium-vaporization events. These data indicate that severely damaged subassemblies will not be coolable in situ by returning sodium after the accident, because of steel and fuel blockages, and suggest that the severely damaged, melted fuel will eventually disperse upward under the influence of vapor generated by decay heating.

- 57 (ANL--77-43) VAPOR-EXPLOSION EXPERIMENTS WITH SUBCOOLED FREON. Henry, R.E.; McUmber, L.M. (Argonne National Lab., IL (USA)). Jun 1977. Contract W-31-109-ENG-38. 32p. Dep. NTIS, PC A03/MF A01.

Vapor-explosion experiments were conducted in a well-wetted Freon-22 and mineral-oil system in which the initial temperature of both the Freon and the mineral oil were varied over a wide range. These experiments were specifically conducted to investigate the importance of interface temperature in determining the explosive behavior of a given system. The results clearly demonstrate that the interface temperature developed upon intimate liquid-liquid contact is a valid characterization of the explosive potential of a given system.

- 58 (ANL--77-44) SUMMARY AND EVALUATION OF FUEL DYNAMICS TRANSIENT-OVERPOWER EXPERIMENTS: STATUS 1974. Deitrich, L.W.; Doerner, R.C.; Hughes, T.H.; Wright, A.E. (Argonne National Lab., IL (USA)). Jun 1977. Contract W-31-109-ENG-38. 128p. Dep. NTIS, PC A07/MF A01.

The report summarizes and evaluates experiments conducted in the Transient Reactor Test Facility (TREAT) using the Mark-II loop facility. The tests discussed are of the E and H series. Detailed descriptions of test conditions and test results as of February 1974 are presented. Since all data have not been acquired on all experiments, this report must be considered interim in nature. Particular emphasis is placed on data relevant to Fast Test Reactor (FTR) safety-analysis efforts.

- 59 (ANL--77-46) FIRST MECHANIZED IN-SERVICE INSPECTION OF EBR-II STEAM GENERATOR. Longua, K.J.; Whitham, G.K.; Reimann, K.J. (Argonne National Lab., IL (USA)). Sep 1977. Contract W-31-109-ENG-38. 39p. AT.

A mechanized ultrasonic test system was developed and built at Southwest Research Institute (SwRI) to conduct the first in-service inspection of the 2 1/4, Cr-1 Mo steel duplex tubing in the EBR-II steam-generator system. This inspection required tube inspection from the bore side, complete volumetric examination, wall-thickness measurement, and evaluation of the braze condition. The test equipment was thoroughly checked and routinely calibrated, using a standard containing artificial flaws. Artificial flaws as small as 1.6 mm long by 0.25 mm deep were readily detected. Forty-eight of 73 tubes were inspected, and 15 of them were rescanned for verification of indications. Calculations showed that the largest indication, 5 mm long by 0.25 mm deep, would not grow to sufficient size before the next scheduled inspection to cause concern. The second indication was classified as a small dent. Before this inspection, an exploratory manual ultrasonic testing examination was conducted on EV-702, but inherent limitations made this test unsuitable for definition of the duplex-tube condition. This test was useful, however, in specifying the requirements for the

mechanized ultrasonic test system. Finally, parallel laboratory studies were conducted to examine effects of flaw size and orientation on ultrasonic indications. While the results of these studies were not directly used for the in-service inspection, they do form a basis for (1) understanding the ultrasonic test signals in duplex 2 1/4 Cr-1 Mo steel tubes and (2) possible future refinements of the mechanized ultrasonic test system built by SwRI.

- 60 (ANL--77-48) PHYSICS OF REACTOR SAFETY. QUARTERLY REPORT, JANUARY--MARCH 1977. (Argonne National Lab., IL (USA)). Jun 1977. Contract W-31-109-ENG-38. 57p. Dep. NTIS, PC A04/MF A01.

This report summarizes work done on reactor safety, Monte Carlo analysis of safety-related critical assembly experiments, and planning of DEMI safety-related critical experiments. Work on reactor core thermal-hydraulics is also included.

- 61 (ANL--77-52) APPLICATION OF THE PULSED-NEUTRON ACTIVATION TECHNIQUE FOR FLOW MEASUREMENT AT EBR-II. Price, C.C.; Sackett, J.I.; Curran, R.N.; Livengood, C.L.; Kehler, P.; Forster, G.A. (Argonne National Lab., IL (USA)). Nov 1977. Contract W-31-109-ENG-38. 69p. AT.

The pulsed-neutron-activation (PNA) flow-measuring technique as applied to in situ fluid-flow measurements at EBR-II is described. Analytic relationships are derived for modeling the process and estimating the uncertainty in the measurement. Results from measurements of both water flow and secondary-sodium flow are presented. Results from PNA measurements on the water side of the EBR-II steam system have led to better definition of plant parameters. Results from sodium-flow measurements are used to provide a correlation for in situ calibration of the electromagnetic sodium flowmeter in the secondary system.

- 62 (ANL--77-53) APPARATUS FOR TESTING SMOOTH AND RIBBED TUBING WITH PULSED EDDY CURRENTS. Sather, A. (Argonne National Lab., IL (USA)). May 1978. Contract W-31-109-ENG-38. 26p. Dep. NTIS, PC A03/MF A01.

The use of outside diameter ribs on nuclear reactor tubing presents severe problems to nondestructive testing of this tubing. A pulsed eddy-current apparatus for inspecting the integrity of smooth and ribbed tubing has been developed at Argonne National Laboratory. Test results for this apparatus and analysis and circuit block diagrams are presented. In addition, transducer and preamplifier features, scanning procedures, and theory of operation are discussed. To date, this apparatus has provided the best solution to the ribbed tubing problem because of its capability to time-sample data.

- 63 (ANL--77-54) FINAL REPORT ON TREAT TEST R3, A SINGLE-PIN LOSS-OF-FLOW EXPERIMENT. Holtz, R.E.; Grohmes, M.A.; Spencer, B.W.; Miller, C.E.; Kramer, N.A.; Testa, F.J.; Murphy, W.F. (Argonne National Lab., IL (USA)). Jul 1977. Contract W-31-109-ENG-38. 57p. Dep. NTIS, PC A04/MF A01.

Test R3 was the first of the R-series fuel-element meltdown tests performed to support modeling and code development for analysis of hypothetical, whole-core accidents in LMFBR's. Test R3 served as a proof test for the subsequent R4--R8 sequence of seven-pin tests examining coolant, cladding, and fuel behavior under thermal and hydraulic conditions

representative of a hypothetical loss-of-coolant flow (LOF) with failure to scram in FFTF. The high heat loss radiated from the single test fuel element run at full power prevented representative prototypicality of the test events. However, the test was significant as a prelude to subsequent seven-pin tests, and also from the standpoint of materials interactions and fuel-motion detection by the hodoscope. Coolant voiding began when the inlet flow rate was down to about 1.7 m/s (approximately 22% of initial flow rate). Upon melting, most of the cladding moved downward, eventually solidifying in the bottom of the test region, forming an inlet blockage. For the conditions of this single-pin test, molten-cladding flooding and upper blockage formation were not found. Subsequent fuel melting was relatively minor; pellets at the top and bottom of the active fuel showed no melting, while about 43% area melting occurred at the hot spot. At no time during the meltdown sequence was there any sodium reentry or significant pressurization events.

- 64 (ANL--77-66) STATISTICAL STUDY OF EBR-II FUEL ELEMENTS MANUFACTURED BY THE COLD LINE AT ARGONNE-WEST AND BY ATOMICS INTERNATIONAL. Harkness, A.L. (Argonne National Lab., IL (USA)). Sep 1977. Contract W-31-109-ENG-38. 24p. Dep. NTIS, PC A02/MF A01.

Nine elements from each batch of fuel elements manufactured for the EBR-II reactor have been analyzed for ²³⁵U content by NDA methods. These values, together with those of the manufacturer, are used to estimate the product variance and the variances of the two measuring methods. These variances are compared with the variances computed from the stipulations of the contract. A method is derived for resolving the several variances into their within-batch and between-batch components. Some of these variance components have also been estimated by independent and more familiar conventional methods for comparison.

- 65 (ANL--77-69) PHYSICS OF REACTOR SAFETY. QUARTERLY REPORT, APRIL--JUNE 1977. (Argonne National Lab., IL (USA)). Sep 1977. Contract W-31-109-ENG-38. 47p. Dep. NTIS, PC A03/MF A01.

The work in the Applied Physics Division includes reports on reactor safety program by members of the Reactor Safety Appraisals Group, Monte Carlo analysis of safety-related critical assembly experiments by members of the Theoretical Fast Reactor Physics Group, and planning of safety-related critical experiments by members of the Zero Power Reactor (ZPR) Planning and Experiments Group. Work on Reactor core thermal-hydraulic code development performed in the Components Technology Division is also included in the report.

- 66 (ANL--77-70) ALTERNATIVE FUEL CYCLE OPTIONS: PERFORMANCE CHARACTERISTICS AND IMPACT ON NUCLEAR POWER GROWTH POTENTIAL. Chang, Y.I.; Till, C.E.; Rudolph, R.R.; Deen, J.R.; King, M.J. (Argonne National Lab., IL (USA)). Sep 1977. Contract W-31-109-ENG-38. 43p. Dep. NTIS, PC A03/MF A01.

The fuel utilization characteristics for LWR, SSCR, CANDU and LMFBR reactor concepts are quantified for various fuel cycle options, including once-through cycles, thorium cycles, and denatured cycles. The implications of various alternative reactor deployment strategies on the long-term nuclear power growth potential are then quantified in terms of the maximum nuclear capacity that can be

achieved and the growth pattern over time, subject to the constraint of a fixed uranium-resource base. The overall objective of this study is to shed light on any large differences in the long-term potential that exist between various alternative reactor/fuel cycle deployment strategies.

- 67 (ANL--77-72) CYCLIC-DEFORMATION RESISTANCE OF WELD-DEPOSITED TYPE 16-8-2 STAINLESS STEEL AT 593°C. Raske, D.T. (Argonne National Lab., IL (USA)). Aug 1977. Contract W-31-109-ENG-38. 37p. AT.

The creep fatigue and cyclic-deformation behavior of Type 16-8-2 (16 percent Cr-8 percent Ni-2 percent Mo) stainless steel weld metal is discussed. Tests were conducted in air at 593°C and a strain rate of $4 \times 10^{-3} \text{ s}^{-1}$. Comparisons with data for Type 316 stainless steel base metal indicate that the weld metal has significantly longer fatigue lives for 360-, 1.8×10^3 -, and 1.8×10^4 -s tension hold-time tests. This longer life is attributed to the fine duplex microstructure of the weld metal that inhibits the growth rate of cracks. Additional studies on the cyclic-deformation behavior of the weld metal indicate that this material is strain-history dependent; therefore, a unique cyclic stress-strain curve does not exist. Monotonic tension tests after cyclic straining result in a different stress-strain curve from that obtained from companion fatigue tests at various completely reversed constant strain ranges. A comparison of the fracture morphology of creep-fatigue and creep-rupture specimens indicates that differences resulting from these tests can be attributed to different failure mechanisms. 32 figures, 5 tables.

- 68 (ANL--77-73) MODEL FOR TURBULENT MOMENTUM AND HEAT TRANSPORT IN LARGE ROD BUNDLES. Sha, W.T.; Launder, B.E. (Argonne National Lab., IL (USA)). Jan 1979. Contract W-31-109-ENG-38. 32p. AT.

A quasi-continuum model for turbulent momentum and heat transport in large rod bundles has been developed. This model has been derived from a subchannel analysis and adapted to a quasi-continuum form by introducing concepts of porosity and distributed resistance. The effects of turbulent kinetic energy generation due to shear, viscosity, diffusion, geometric effects, buoyancy, and Reynolds number are explicitly included. The proposed model of turbulence is relatively simple, yet it is believed to provide a framework for taking account of important turbulent mechanisms in rod bundles.

- 69 (ANL--77-74) BILINEAR CYCLIC STRESS-STRAIN ANALYSIS FOR INCOLOY 800. Maiya, P.S. (Argonne National Lab., IL (USA)). Nov 1977. Contract W-31-109-ENG-38. 27p. AT.

This report describes the bilinear stress-strain analysis under cyclic loading conditions for the alloy Incoloy 800. Although the method for determining the bilinear stress-strain parameters is based on a procedure proposed in the RDT Standard F9-1 for inelastic analysis of Fast Flux Test Facility (FFTF) components, the accuracy and consistency of results have been improved by an analytical technique, which also resulted in certain simplifications. The bilinear stress-strain parameters of solution-annealed Incoloy 800 (Heat HH7058A) under cyclic loading conditions at a strain rate ($\epsilon/\text{sub } t$) of $8.6 \times 10^{-5} \text{ s}^{-1}$, total strain range ($\Delta\epsilon/\text{sub } t$) of 0.2 to 0.8 percent, and temperatures of room temperature to 593°C (1100°F) have been determined. The

dependence of bilinear parameters on temperature and strain is discussed. The cyclic-hardening characteristics based on correlation of yield parameter k with accumulated plastic strain are also presented.

- 70 (ANL--77-76) ANALYSIS OF EBR-II LOW-POWER DOSIMETRY RUN 78C. Franklin, F.C.; Ebersole, E.R.; Heinrich, R.R. (Argonne National Lab., IL (USA)). Dec 1977. Contract W-31-109-ENG-38. 49p. AT.

This report compares calculated reaction rates based on neutron-transport calculations in RZ and XY geometries with measured values from a low-power dosimetry test in EBR-II. Axial distributions of ^{235}U and ^{238}U fission rates and ^{238}U capture rate are given for various radial locations along the length of the core and the axial reflectors, and along the length of the radial steel reflectors. Reaction rates, primarily at the reactor midplane, are given for a number of fission and capture reactions. The analytical RZ- and XY-geometry models used for the neutronics calculations are described.

- 71 (ANL--77-77) PLANE-STRAIN STRESS INTENSITY FACTORS FOR CRACKED HEXAGONAL SUBASSEMBLY DUCTS. Petroski, H.J.; Glazik, J.L.; Achenbach, J.D. (Argonne National Lab., IL (USA)). Oct 1977. Contract W-31-109-ENG-38. 44p. Dep. NTIS, PC A03/MF A01.

Plane-strain stress intensity factors for a pressurized hexagonal subassembly duct with a crack in a corner or midflat are presented in convenient graphical form for representative LMFBR hexcan dimensions. Corner-crack calibrations based on several different models of the round hexcan corner are determined first in order to bound the stress intensity factor. A subsequent finite-element analysis of a uniformly pressurized hexcan with a corner crack gives accurate data for the stress intensity factor from which a weight function for this geometry may be constructed. The effects of different numbers of cracks, different locations for cracks, and different loading modes are discussed briefly, and some comments are made on the application of linear elastic fracture mechanics to cracked hexagonal ducts that have suffered a high degree of irradiation embrittlement.

- 72 (ANL--77-82) COMPUTATION OF THE WEIGHT FUNCTION FROM A STRESS INTENSITY FACTOR. Petroski, H.J.; Achenbach, J.D. (Argonne National Lab., IL (USA)). Oct 1977. Contract W-31-109-ENG-38. 28p. Dep. NTIS (US Sales Only), PC A03/MF A01.

A simple representation for the crack-face displacement is used to compute a weight function solely from stress intensity factors for a reference loading configuration. Crack-face displacements given by the representation are shown to be in good agreement with analytical results for cracked tensile strips, and stress intensity factors computed from the weight function agree well with those for edge cracks in half-planes, radial cracks from circular holes, and radially cracked rings. The technique involves only simple quadrature, and its efficacy is demonstrated by the example computations. The weight function for a corner crack in an LMFBR hexagonal subassembly duct is constructed from stress-intensity-factor results for the uniformly overpressurized case, and it is shown how this may be used to determine the stress intensity factors for other loading cases.

- 73 (ANL--77-86) INVESTIGATIONS OF MATERIALS COMPATIBILITY RELEVANT TO THE EBR-II SYSTEM: FY 1977 AND THE TRANSITION PERIOD. Longua, K.J.; Ruther, W.E.; Shields, J.A.; Olson, W.H.; Clark, A.F. (Argonne National Lab., IL (USA)). Feb 1978. Contract W-31-109-ENG-38. 30p. AT.

The failure of newly designed thermocouple assemblies in the steam system was caused by severe corrosion due to residual brazing flux left inside the protective tube. Examination of a failed bellows-and-pivot-rod assembly inside the turbine initial-pressure-regulation assembly revealed intergranular cracking in the heat-affected zone at the root convolution of the welded bellows. The annual inspection of the steam plant included an in-service type of inspection. Burst tests of fuel-element tubes were performed in irradiated and unirradiated defected hexagonal subassembly ducts, using both single and multiple defected tubes to simulate fuel-element failure. A fabric to be used in an enclosure to protect the EBR-II fuel transfer arm was tested successfully in sodium at 135 to 200°C. Thermal-shock testing of stainless steel-clad thermocouples at rates of 2600°C/s confirmed their capability to withstand sudden changes in temperature. Out-of-reactor experiments indicated that graphite and reticulated vitreous carbon will trap ^{137}Cs and are compatible with 371°C sodium. Experiments were performed with thermal-expansion-difference temperature monitors to determine the extent of stress relaxation during calibration.

- 74 (ANL--77-87) SUMMARY, ANALYSIS, AND COMPARISON OF DATA FROM EXPERIMENTS WITH THE EBR-II MARK IIB OSCILLATOR ROD. Larson, H.A.; Dean, E.M. (Argonne National Lab., IL (USA)). Apr 1978. Contract W-31-109-ENG-38. 51p. AT.

The Mark IIB oscillator rod was used in EBR-II runs 48-55. Data were collected, digitized, and processed by on-line computer analysis, and Fourier analysis was used to obtain frequency-response functions. Feedback frequency-response functions are compared with those obtained by rod-drop experiments and are qualitatively identical. Experimental results are also compared with the EBR-II dynamic-simulation computer program, EROS. Through use of basic neutronic and heat-transfer principles, this program is shown to adequately represent the reactor feedback frequency-response function. The experimental data from EBR-II runs 48-55 are presented in both tabular and graphic form.

- 75 (ANL--77-88) APPROXIMATIONS OF GAMMA CROSS SECTIONS FOR FAST NUCLEAR REACTORS. Grimm, K.N.; Meneghetti, D. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 24p. Dep. NTIS, PC A02/MF A01.

The report shows a method to approximate a P_1 scattering solution for the flux in a fast reactor, using an isotropic, but not a diagonal-transport-approximation scattering matrix. Presented are flux errors relative to a P_1 solution for different levels of transport approximation in an EBR-II type of core surrounded by a stainless steel reflector. Problems associated with the use of the method are also presented.

- 76 (ANL--77-91) COUPLED HYDRODYNAMIC-STRUCTURAL RESPONSE ANALYSIS OF PIPING SYSTEMS. A-Moneim, M.T. (Argonne National Lab., IL (USA)). Feb 1978. Contract W-31-109-ENG-38. 88p. AT.

The report describes in detail the expansion of the ICEPEL code for the coupled hydrodynamic-

structural response analysis of pipe-elbow loops to general piping systems. A generalized piping-component model, a branching tee junction, and a surge-tank model are introduced and coupled with the pipe-elbow loop model so that a general piping system under the effect of internally traveling pressure pulses can be analyzed hydrodynamically, as well as structurally. Optional capabilities are included in the generalized piping-component model so that it can be used in modeling pipe-flow area changes, valves, and heat exchangers. The options are: nine different axisymmetric exterior wall configurations, elastic-plastic exterior walls, interior rigid-wall simulation, and heat-exchanger tube-bundle model. The Implicit Continuous-Fluid Eulerian (ICE) differencing technique is used in the hydrodynamic part of the analysis; the Waves in Hysteretic Arbitrary Media (WHAM) finite-element scheme is used in the structural response part of the analysis. Sample problems that demonstrate the capabilities and limitations of the method are presented. The results are discussed and conclusions are made in connection with further development of the method. Detailed input-output instructions for the ICEPEL code use are also presented.

- 77 (ANL--77-92) PHYSICS OF REACTOR SAFETY. QUARTERLY REPORT, JULY--SEPTEMBER 1977. (Argonne National Lab., IL (USA)). Dec 1977. Contract W-31-109-ENG-38. 40p. Dep. NTIS, PC A03/MF A01.

The work in the Applied Physics Division includes reports on reactor safety program by members of the Reactor Safety Appraisals Group, Monte Carlo analysis of safety-related critical assembly experiments by members of the Theoretical Fast Reactor Physics Group, and planning of safety-related (ZPR) Planning and Experiments Group. Work on reactor core thermal-hydraulic code development performed in the Components Technology Division is also included.

- 78 (ANL--77-93) TRANSIENT-OVERPOWER TEST E8 ON FFTF-TYPE LOW-POWER-IRRADIATED FUEL. Simms, R.; Lo, R.K.; Murphy, W.F.; Stanford, G.S.; Rothman, A.B. (Argonne National Lab., IL (USA)). Dec 1977. Contract W-31-109-ENG-38. 124p. Dep. NTIS, PC A06/MF A01.

Test E8 simulated a hypothetical \$3/s transient overpower accident in an LMFBR using seven (Pu, U)O₂ fuel elements of the FTR type. The test elements were preirradiated in the PNL-10 assembly in EBR-II to 5 at. % burnup at 30 kW/m. The preirradiation in EBR-II caused a fuel-restructuring range characteristic of a low-to-moderate power microstructure for FTR. Failure predictions indicated that fuel with this microstructural characteristic would fail at a lower energy deposition than fuel irradiated at higher power. Data from test-vehicle sensors, hodoscope, and postmortem examinations were used to construct the sequence of events occurring within the test zone. The sequence occurred incoherently across the test cluster, the initial event occurring abruptly at about 29 times nominal power level at an estimated stored energy of about 925 kJ/kg with 50% of the fuel above the solidus at the suspected failure site. After the initial failure, about 2% of the total mass of test fuel was ejected above the original top of the active fuel column. Sodium voiding occurred rapidly. A fuel-debris blockage also apparently prevented further fuel dispersal. Inherent test-vehicle limitations, loss of flow-tube geometry, and nontypical power generation after failure may have caused a departure from the fuel motion predicted for FTR conditions. No

violent fuel-coolant interaction or associated work-energy conversion to the coolant was observed.

- 79 (ANL--77-94) TEST E3 ON HIGH-ENERGY TRANSIENT MELTDOWN OF IRRADIATED UO_2 IN A TREAT MARK-II LOOP. Deitrich, L.W.; Dickerman, C.E.; Willis, F.L. (Argonne National Lab., IL (USA)). Dec 1977. Contract W-31-109-ENG-38. 53p. Dep. NTIS, PC A04/MF A61.

Three "gassy" irradiated UO_2 pins were run past failure in a 35-ms natural TREAT transient in TREAT loop Test E3 to guide in choosing between two modeling assumptions regarding energy conversion from sodium vaporization in an irradiated-oxide-fuel/coolant interaction (FCI): (1) The trapped fission gas enhances fuel fragmentation upon failure, accelerates fuel-coolant mixing, and thus promotes an energetic FCI, or, conversely, (2) the trapped fission gas blankets the fuel and reduces the rate of heat transfer to sodium, preventing an energetic FCI. The sample pins were run in the Materials Testing Reactor (MTR) to burnup levels of 4.8, 4.8, and 6.3 at. percent. Gas retention of 77 percent was estimated, based on sibling-pin data. Each pin consisted of a short fuel stack about 145 mm (5.7 in.) long, clad with a 4.42-mm (0.174-in.)-OD, annealed Type 304 stainless steel jacket, 0.23 mm (0.009 in.) thick. Failure occurred before coolant boiling at a fuel energy input in the highest energy location of the hottest pin of 857 kJ/kg within 5 ms of inception of melting. Failure was marked by sharp pressure pulses and sodium ejection. The maximum growth rate of void volume was about $9.5 \times 10^{-3} \text{ m}^3/\text{s}$ the largest recorded in a TREAT loop test. The initial pressure pulses recorded at inlet and outlet were 5.9 and 3.3 MPa, respectively. Although the work was the highest for all the TREAT loop tests, it was only about 4 kJ/kg, of which about half could be accounted for by expansion of fission gas. There was no evidence for an energetic FCI of the type that could be calculated on the basis of rapid fuel-coolant mixing and heat transfer.

- 80 (ANL--77-95) CLADDING FAILURE BY LOCAL PLASTIC INSTABILITY. Kramer, J.M.; Deitrich, L.W. (Argonne National Lab., IL (USA)). Dec 1977. Contract W-31-109-ENG-38. 36p. Dep. NTIS, PC A03/MF A01.

Cladding failure is one of the major considerations in analysis of fuel-pin behavior during hypothetical accident transients since time, location, and nature of failure govern the early postfailure material motion and reactivity feedback. Out-of-pile thermal transient tests of both irradiated and unirradiated fast-reactor cladding show that local plastic instability, or bulging, often precedes rupture and that the extent of local instability limits the initial rip length. To investigate the details of bulge formation and growth, a perturbation analysis of the equations governing large deformation of a cylindrical shell has been developed, resulting in a set of linear differential equations for the bulge geometry. These equations have been solved along with appropriate constitutive equations and various constraints on the ends of the cladding. Sources for bulge formation that have been considered include initial geometric imperfections and thermal perturbations due to either eccentric fuel pellets or nonsymmetric cooling. Of these, only the first is relevant to out-of-pile burst tests. Here it has been found that the most likely imperfection that will grow unstably to failure leads to a bulge around half the circumference with an axial length 1.1 times

the deformed diameter. This is in general agreement with burst-test results. For the case of in-reactor fuel pins, it has been found that thermal perturbations can significantly affect local instability, particularly if the deformation process is thermally activated with a high activation energy.

- 81 (ANL--77-XX-62, pp 24-27) INTERNATIONAL FAST BREEDER REACTOR DEVELOPMENT. Yevick, J.G. 1977.

From 16. nuclear engineering education conference on international nuclear engineering - development and planning; Argonne, IL, USA (28 Feb 1977).

The LMFBR development programs in the U.S., France, German Federal Republic, Italy, Japan, U.K., and the U.S.S.R. are reviewed.

- 82 (ANL--78-4) PTAC: A COMPUTER PROGRAM FOR PRESSURE-TRANSIENT ANALYSIS, INCLUDING THE EFFECTS OF CAVITATION. Kot, C.A.; Youngdahl, C.K. (Argonne National Lab., IL (USA)). Sep 1978. Contract W-31-109-ENG-38. 95p. AT.

PTAC was developed to predict pressure transients in nuclear-power-plant piping systems in which the possibility of cavitation must be considered. The program performs linear or nonlinear fluid-hammer calculations, using a fixed-grid method-of-characteristics solution procedure. In addition to pipe friction and elasticity, the program can treat a variety of flow components, pipe junctions, and boundary conditions, including arbitrary pressure sources and a sodium/water reactor. Essential features of transient cavitation are modeled by a modified column-separation technique. Comparisons of calculated results with available experimental data, for a simple piping arrangement, show good agreement and provide validation of the computational cavitation model. Calculations for a variety of piping networks, containing either liquid sodium or water, demonstrate the versatility of PTAC and clearly show that neglecting cavitation leads to erroneous predictions of pressure-time histories.

- 83 (ANL--78-9) EBR-II ENVIRONMENTAL INSTRUMENTED SUBASSEMBLY XX08: ENGINEERING AND ASSEMBLY. Smaardyk, A.; Filewicz, E.C.; Longnecker, A.A.; Poloncsik, J.; Tokar, J.V.; Walker, D.E.; Wendte, J.C. (Argonne National Lab., IL (USA)). May 1978. Contract W-31-109-ENG-38. 72p. AT.

Subassembly XX08 is a fueled and instrumented subassembly designed primarily for an ongoing program to investigate the thermal-hydraulic core environment within EBR-II under normal and off-normal plant operating conditions. The Mark-II fuel is expected to provide XX08 with an irradiation lifetime three times as great as that attained with XX07, i.e., 9 versus 2.9 at. percent burnup. A burnup of 9 at. percent is equivalent to about 29,000 Mwt days of EBR-II reactor operation, which corresponds to 11 reactor runs at 2700 Mwd per run. Instrumentation within XX08 includes 16 spacer-wire coolant thermocouples with junctions at various axial and radial locations within the fuel-element bundle, six top-of-core-elevation fuel-pin thermocouples, two subassembly-outlet-coolant thermocouples, two permanent-magnet flowmeters for measuring subassembly flow, and two rhodium self-powered detectors for neutron-flux determinations. Signals from the instrumentation are processed and recorded using the EBR-II digital data-acquisition system (DAS). This report provides detailed information on engineering, design, and assembly of XX08.

- 84 (ANL--78-10, pp 192-201) STATUS OF THE POST ACCIDENT HEAT REMOVAL PROGRAM AT GFK FOR FAST BREEDER REACTORS. Hofmann, F.; Alsmeyer, H.; Hame, W.; Perinic, D.; Schulz, B. (Gesellschaft fuer Kernforschung, Karlsruhe, Ger.). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

Long term coolability and stable containment of irradiated core material with internal decay heat generation must be possible under quite a number of circumstances of various accident conditions in a LMFBR. The post accident heat removal (PAHR) assessment has to assure coolability for situations after both local and global core disruptions. In-vessel and on a higher level of defense even ex-vessel coolability of core material has to be established. The present state of post accident heat removal R&D activities at GfK is summarized.

- 85 (ANL--78-10, pp 5-8) ROLE OF MECHANISTIC ANALYSIS IN PROVIDING PAHR INITIAL CONDITIONS. Jackson, J.F. (Los Alamos Scientific Lab., NM). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

The objective of this paper is to discuss how Postaccident Heat Removal (PAHR) calculations interface with the earlier accident phases in overall mechanistic accident analyses, to briefly review the evolution of the transition phase as an important path in LMFBR core disruptive accidents, and to assess the impact of the transition phase on PAHR analyses.

- 86 (ANL--78-10, pp 9-14) VIEWS ON POST-ACCIDENT HEAT REMOVAL. Marchese, A.R. (Nuclear Regulatory Commission, Washington, DC). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

Some thoughts on the technical needs, fundamental direction, and relative emphasis that should be characteristic of studies on post-accident heat removal are presented.

- 87 (ANL--78-10, pp 26-34) CURRENT STATE OF ANALYSIS OF RECRITICALITY UNDER PAHR CONDITIONS. Abramson, P.B. (Argonne National Lab., IL). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

In the highly unlikely event of failure to scram following a loss of coolant flow in an LMFBR, fuel pins would melt and fuel, cladding, and coolant would be redistributed. The redistribution of materials is generically the most difficult of the reactor safety analysis problems. Since as materials relocate, their hydrodynamic and thermodynamic interactions are altered, any error in the calculation of the relocations is rapidly compounded to a point that what is calculated may have very little resemblance to reality. Attempts to deterministically calculate these material interactions and motions have taken on a parametric nature in an effort to provide a reasonable set of bounds on the macroscopic accident consequences (such as the potential challenge to the containment). Unfortunately, while some efforts have been fruitful, many of these parametric attempts have resulted in sets

of bounds which are so wide as to provide no useful information. Because each material in the reactor has specific effects upon the neutronic behavior, and because a typical LMFBR contains so much fissionable material that a relatively small fraction of the original core can itself cause a prompt critical configuration, and because the effects of a neutronic prompt critical excursion are felt neither hydrodynamically nor thermodynamically until the configuration is reached, we do not have the ability to rule out by deterministic analysis the occurrence of second prompt critical excursion (recriticality). The current state of analytical efforts to understand recriticality is reviewed.

- 88 (ANL--78-10, pp 35-46) SECONDARY CRITICALITY EVALUATIONS OF POSTULATED CORE MELT-DOWN ACCIDENTS IN EARLY-SIZED LMFBR'S. Harless, W.H.; Talwar, B.; Gluekler, E.L. (General Electric Co., Sunnyvale, CA). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

Secondary criticality analyses have been performed for an early-sized LMFBR. The results indicate that recriticality will occur only as a result of extremely unlikely combinations of failures which lead to melt-down of a large portion of the reactor core, subsequent formation of a relatively pure fuel layer, and/or substantial vessel deformation. Design features were identified that could be used to eliminate recriticality or mitigate its effects if needed to meet safety requirements. The presently existing large uncertainties in the containment response and radiological consequences of single or multiple recriticalities may be reduced with the aid of dynamic analyses that couple neutronic, thermal, and hydrodynamic methods.

- 89 (ANL--78-10, pp 49-60) PRELIMINARY RESULTS FROM INITIAL IN-PILE DEBRIS BED EXPERIMENTS. Rivard, J.B. (Sandia Labs., Albuquerque, NM). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

The initial series of three experiments in the ACPR, designated D-1, D-2, and D-3, was designed to provide basic data on the cooling capacity of representative debris beds under typical accident conditions. Important issues addressed include: (1) whether or not bed depth is an influencing parameter on cooling capacity; (2) what the influence of sodium subcooling is; and (3) how the results correlate with the results of nonreactor experiments and with analytical treatments. Besides providing this frame of reference for safety assessment, it was expected that an initial understanding of bed phenomena would be gained, the details of which could be determined in future experiments. This paper reports preliminary results from the D-series of three experiments.

- 90 (ANL--78-10, pp 62-72) EFFECT OF PARTICLE SIZE DISTRIBUTION AND NON-HEATED CONSTITUENTS ON DEBRIS BED DRY-OUT. Dhir, V.K.; Catton, I. (Univ. of California, Los Angeles). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

Past studies of debris bed dry-out have been made using beds of a single particle size (UCLA) or a particle size distribution that attempted simulation of TREAT tests or ANL experiments (ANL, SANDIA). A systematic

variation of the particle size distribution has not yet been made. It is the purpose of this work to report on a study of the effect of particle size distribution on debris bed dry-out. Another aspect of this work considered the effect of mixtures of unheated and heated constituents on dry-out. Measurements were also made of segregated beds. The dry-out heat flux data are compared with models based on semi-theoretical correlations developed earlier for deep beds.

- 91 (ANL--78-10, pp 74-80) PARTICULATE BED FORMATION AND PHENOMENA. Gabor, J.D.; Sowa, E.S.; Pavlik, J.R.; Cassulo, J.C. (Argonne National Lab., IL). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

The mechanics of particulate debris formation by interaction of molten-core material and sodium and the subsequent phenomena of sodium boiling within the interstices of the particulate debris are important elements in Post-accident Heat Removal analyses. Recent large-scale particulate-bed-formation tests and investigations on channel formation and hydraulic phenomenon associated with heat removal from fragmented core debris are reported here.

- 92 (ANL--78-10, pp 81-87) EXPERIMENTAL STUDIES OF A HEAT GENERATING POROUS BED. Lee, D.O.; Nilson, R.H. (Sandia Labs., Albuquerque, NM). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

The paper describes experimental observations and qualitative interpretations of flow patterns and temperature distributions for a series of steady state bench experiments in which fission product decay is simulated by Joule heating of the liquid phase. Both single-phase and two-phase boiling phenomena are examined. Particular emphasis is given to the influence of subcooling at the upper surface of a boiling bed. In addition, a brief discussion of similitude parameters and experimental scaling is presented. These considerations suggest a good degree of similarity between the D2 series of in pile experiments at Sandia and the supportive simulant experiments which are reported here.

- 93 (ANL--78-10, pp 92-98) CORE DEBRIS DEPOSITION IN THE PRIMARY HEAT TRANSPORT SYSTEM PIPING OF LMFBRs. Tsai, S.S. (Brookhaven National Lab., Upton, NY). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

A hypothetical core disruptive accident may lead to the upward ejection of core fuel into the upper sodium plenum of a liquid metal fast breeder reactor (LMFBR). Subsequently, the fuel will quench and fragment in the subcooled sodium. Due to the gravitational force, some of the core debris may settle onto horizontal surfaces in the vessel, and the remainder of the debris will be swept out of the reactor vessel into the primary heat transport system piping. Depending on the sodium flow rate in the primary system piping, the debris will either remain suspended or settle out of the sodium stream. It is assumed that the three primary loops are intact and sodium flow is maintained by the pony motors. Due to the low pony-motor-driven sodium flow rate, the core debris is expected to settle out of the sodium

stream and deposit on the bottom walls of the primary piping. It is the purpose of this study to investigate the possibility of forming a debris bed on the piping walls and to assess the coolability of such a bed.

- 94 (ANL--78-10, pp 205-214) EXPERIMENTAL AND THEORETICAL STUDIES ON WATER AND GAS RELEASE FROM HEATED CONCRETE. McCormack, J.D.; Postma, A.K. (Hanford Engineering Development Lab., Richland, WA). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

Procedures currently used in licensing of nuclear facilities require analysis of postulated accidents which are more severe than "design basis" events. For breeder program reactors, some accidents in this severe category involve spillage of sodium coolant onto concrete protected by a steel liner. Heat transfer through the liner heats the concrete, causing a part of the mixing water and other gases to be driven off. These gases would add to pressure in the containment atmosphere, and the water vapor can form hydrogen if it contacts sodium. Evaluations of containment integrity for such postulated accidents will be aided by knowledge of how much water and other gases are released from heated concrete. Results are presented of a research effort at Hanford Engineering Development Laboratory designed to improve understanding of the release of water and gases from heated concrete.

- 95 (ANL--78-10, pp 223-232) CORE DEBRIS PENETRATION INTO CONCRETE. Baker, L. Jr.; Cheung, F.B.; Bingle, J.D. (Argonne National Lab., IL). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

Analyses of the behavior of core debris following an HCDA require a thorough understanding of the interactions of core debris with concrete. The nature of the attack of core debris on the concrete below the reactor cavity is important because it involves the final line of defense against violation of the containment. One of the most important factors to be determined is the ratio of sideward to downward growth of molten core debris pools in concrete. Such a determination requires evaluations of both the "concrete-side" heat transfer and gas release behavior and the "pool-side" heat transfer behavior as well as of the coupling between the two. This paper describes an empirical approach to evaluate the "concrete-side" conditions and outlines a preliminary pool-growth model (GROWS II) which uses this approach. Two sets of representative "pool-side" conditions were used for illustrative purposes. No coupling effects such as the changes in pool heat transfer caused by the gases released from the concrete were considered in these preliminary calculations.

- 96 (ANL--78-10, pp 233-242) CONCRETE FAILURE MODES AT ELEVATED TEMPERATURES. Gluekier, E.L. (General Electric Co., Sunnyvale, CA); Dayan, A. 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

Local failure modes of concrete at high surface temperatures have been determined based on microstructural investigation and failure surface calculations. Preliminary results indicate that concrete crushing does not occur

and spallation would occur only under extreme conditions.

- 97 (ANL--78-10, pp 243-249) LARGE SCALE SODIUM TEST FACILITY AT SANDIA LABORATORIES. King, D.L.; Smaardyk, J.E.; Sutherland, H.J.; Sallach, R.A. (Sandia Labs., Albuquerque, NM). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

The large scale sodium test facility in use at Sandia Laboratories was designed to be used for a wide range of sodium interaction experiments. Of initial concern is the interaction of hot sodium with concrete under conditions similar to accident conditions in any sodium cooled reactor. The tests to be run cover the cases of sodium spills and sprays on bare concrete and on cells with defective steel liners. The initial series of tests in progress at the facility involves rapidly dropping from 20 kg to 225 kg of sodium at temperatures from 550°C to 850°C into concrete crucibles that are representative of the concrete to be used at the Clinch River Breeder Reactor Plant (CRBRP). Specific objectives are to obtain sufficient information to completely describe the sodium/concrete interaction including: reaction rate, heat of reaction, gas evolution and flow, movement of the reaction front, reaction product formation, and the effects of variation of drop temperature and reaction area geometries.

- 98 (ANL--78-10, pp 250-257) LARGE SCALE SODIUM INTERACTIONS WITH LIMESTONE CONCRETE. Sallach, R.A.; Smaardyk, J.E.; Sutherland, H.J.; King, D.L.; Dahlgren, D.A. (Sandia Labs., Albuquerque, NM). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

For sodium cooled reactors, knowledge of the interaction of hot sodium with cell liners and, given either a failed liner or a core disruptive accident, the interaction of hot sodium with concrete are necessary for a safety assessment. To better understand what happens when hot sodium contacts concrete, a series of tests is being carried out under conditions which are similar to postulated reactor accident conditions. Specific objectives are to obtain a complete description of the sodium/concrete interaction including heat balance, gas evolution and flow, movement of and heat generation in the reaction zone, reaction product formation, and the layering and movement of the reaction products. The experimental results of the initial three tests of the test series are presented. These tests were "uninstrumented" in that the full array of instrumentation had not yet been installed. The test results do include thermal data from thermocouples immersed in the sodium pool, post-test measurements of penetration depths, and post-test chemical analyses of the reaction products.

- 99 (ANL--78-10, pp 258-266) INTERMEDIATE SCALE SODIUM-CONCRETE REACTION TESTS. Hassberger, J.A. (Hanford Engineering Development Lab., Richland, WA). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

Twelve intermediate scale, sodium-concrete reaction tests have been completed. These tests involved spilling hot (1250 to 1600°F, 677 to 871°C) sodium onto 1 ft³ (.028 m³) specimens of magnetite of limestone concretes. The resulting

reactions were allowed to continue for from 2 to 24 hours, at which time the sodium was decanted from the test article. Included are discussions of the observed penetration depths, chemical analyses of the reaction products, characterization of the evolved gases and a brief summary of laboratory scale experiments performed earlier.

- 100 (ANL--78-10, pp 267-272) HYDROGEN JET RECOMBINATION UNDER POSTULATED BREEDER REACTOR ACCIDENT CONDITIONS. Wierman, R.W. (Hanford Engineering Development Lab., Richland, WA). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

Certain conditions may be postulated in breeder reactor risk assessments for which the potential of hydrogen release to the reactor containment building needs to be evaluated. One of these conditions is a postulated core melt-through accident followed by cavity liner failure allowing sodium to react with the structural concrete resulting in the release of free hydrogen. If the hydrogen-sodium vapor mixtures in the reactor cavity were to jet into the containment building and inherently self-ignite at the source where they enter the reactor containment building, no immediate hydrogen accumulation would occur. However, as the oxygen in the sealed containment building is consumed by hydrogen and sodium, the oxygen concentration may decrease to a point at which the hydrogen may cease burning. The self-ignition and extinguishment characteristics of hydrogen-sodium jets need to be understood for complete breeder reactor safety analysis. Therefore, experiments have been conducted investigating hydrogen jet burning characteristics under conditions similar to those postulated for core melt-through accidents. The data presented below describe the conditions for jet self-ignition and extinguishment using a small-scale apparatus.

- 101 (ANL--78-10, pp 275-285) HEAT TRANSFER BY NATURAL CONVECTION IN CONFINED MEDIUM. Grand, D.; Martin, R.; Tenchine, D. (CEN, Grenoble, France). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

In order to estimate the performances of a core-catcher located inside the main vessel of a pool type reactor, one needs to know the heat transfer coefficient between the core-catcher and the surrounding sodium and also the thermal resistance of the large volumes of sodium between the heat source and the outer ultimate cooling circuit. The development of natural convection in the different volumes of sodium separated by metal walls is also of interest.

- 102 (ANL--78-10, pp 286-296) METHOD FOR ESTIMATING THE PERFORMANCES OF AN INTERNAL CORE-CATCHER IN A POOL-TYPE LMFBR. Kayser, G.; LeRigoleur, C. (Centre d'Etudes Nucleaires, Cadarache, France). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

A large pool-type fast reactor presents two important advantages in concern with PAHR: an important sodium volume with a high heat capacity; and a large diameter allowing the retainment of molten fuel in a very thin layer. The objective of this study is to show the practical interest of these advantages and the possible performances of an internal core catcher.

- 103 (ANL--78-10, pp 303-308) DYNAMIC BEHAVIOR OF SNR-300 CORE RETENTION DEVICE AND EXPERIMENTAL SUPPORT OF THE DESIGN CONCEPT. Friedrich, H.J. (Internationale Atomreaktorbau GmbH, Bergisch Gladbach, Ger.). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

Methods and results are presented of temperature course investigations relating to the large Na-pool underneath the reactor vessel after a hypothetical meltthrough of the core debris. Temperature behavior of this pool controls the thermal load of the containment, steam release, aerosol production, and the thermal load of the core catcher components themselves. The SNR-300 Primary System is designed to withstand the consequences of a HDCA without major leakage of sodium. Since there is a good chance to remove the decay heat either by natural convection through the three primary loops or by the six especially protected emergency coolers, no serious rise of the mean sodium temperature can occur.

- 104 (ANL--78-10, pp 309-316) INVESTIGATION OF MARGINS IN FFTF CONTAINMENT DESIGN. Stepniewski, D.D.; Simpson, D.E.; Peak, R.D. (Hanford Engineering Development Lab., Richland, WA). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

Analyses are presented of predicted phenomena and containment building transients for postulated failures of post-accident heat removal (PAHR) that follow a hypothetical core disruptive accident (HCDA) in the Fast Flux Test Facility (FFTF). The results are included of two cases which typify two classes of failure modes of in-vessel PAHR for FFTF. As previously discussed, a disrupted core would be predicted to remain within the reactor vessel, coolable as a debris bed by the primary HTS loops. There are three independent primary loops; any one loop has sufficient capability to cool the core. If, however, all three loops failed, then the sodium coolant would heat up, reach its boiling point, and eventually boil away into the Reactor Containment Building (RCB). This scenario forms the basis for the 'in-vessel' cases treated in subsequent sections of the paper. On the other hand, if for any reason HCDA core debris were to assume a noncoolable configuration, the fuel could melt its way through successive barriers consisting of the core shield-orifice assemblies, core basket region, and lower dome of the core support structure. If a coolable configuration did not develop in the reactor vessel inlet plenum, the lower vessel head would be penetrated, with molten material release to the guard vessel. It is probable that fuel fragmentation would occur in the guard vessel, resulting in a coolable configuration, a conclusion supported by the ANL M-5 test. If further progression occurred, with penetration of the guard vessel, sodium and core materials would then be released to the reactor cavity floor. This scenario is the basis for 'ex-vessel' cases treated in this paper.

- 105 (ANL--78-10, pp 317-326) EVALUATIONS OF CRBRP THERMAL MARGIN BEYOND THE DESIGN BASE. Schornhorst, J.R.; Strawbridge, L.E.; Bradbury, P. (Westinghouse Advanced Reactors Div., Madison, PA). 1977.

From 3. post-accident heat removal

information exchange; Argonne, IL, USA (2 Nov 1977).

This paper assesses the Thermal Margin Beyond the Design Base (TMBDB). Specifically, the consequences of a whole core melt, resulting in penetration of the reactor vessel and guard vessel are assessed. The plant features provide thermal margin such that containment integrity can be maintained for not less than 24 hours after initiation of the event as required by the Nuclear Regulatory Commission. The design features that provide the thermal margin beyond the design base described are illustrated. Some of these features are provided specifically for TMBDB while other features have augmented capabilities to provide TMBDB.

- 106 (ANL--78-10, pp 327-334) EX-VESSEL CONTAINMENT RESPONSE TO A CORE MELTDOWN. Pyun, J.J.; Gasser, R.D.; Pratt, W.T.; Bari, R.A. (Brookhaven National Lab., Upton, NY). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

The Third Level Thermal Margin (TLM) design analysis for the Clinch River Breeder Reactor (CRBR) assumes (in the extremely unlikely event of a hypothetical core disruptive accident) that melt-through of the reactor vessel and guard vessel will occur 1000 seconds after shutdown. All of the active core and lower axial blanket together with 4.5×10^6 gm of steel is then distributed over the 1219.2 cm diameter reactor cavity floor. Approximately 544×10^6 gm of sodium at 1050°F is also released into the cavity, and it is assumed that the fuel and steel form a sodium-flooded, self-leveling debris bed. As a consequence of this assumption, all of the decay heat is transferred to the sodium. It is also assumed that the steel liner on the reactor cavity floor fails and is followed by a reaction of sodium with concrete. The purpose of the present investigation is to consider the possibility that a molten fuel and steel pool may exist rather than a sodium-flooded debris bed.

- 107 (ANL--78-10, pp 335-342) REVIEW OF CONTAINMENT TRANSIENTS IN A 2500 MW/SUB T/ SODIUM COOLED BREEDER REACTOR FOLLOWING POSTULATED CORE DISRUPTIVE ACCIDENTS. Armstrong, G.R.; Seeman, S.E. (Hanford Engineering Development Lab., Richland, WA). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

Three types of containment configurations for breeder reactors of the 2500 MW(t) class were compared to determine the typical responses associated with each type to selected hypothetical accident conditions. The study was conducted for ERDA-RDD by HEDL to provide insight as to the sensitivity of accident consequences to various design and phenomenological assumptions, and thereby provide potentially useful input to guidance of R and D activities. The types of containment systems investigated in the study were the single, double and containment-confinement designs. Two types of hypothetical accidents were arbitrarily assumed as bases for the initial conditions in the evaluation--the energetic core disruptive accident and the whole core melt-through accident. More than 20 separate cases were analyzed in which selected design and phenomenological assumptions were varied. The primary computer codes used in the study were CACECO, HAA, and COMRADEX.

- 108 (ANL--78-10, pp 353-360) HEAT TRANSFER IN VOLUMETRICALLY-HEATED POOLS WITH GAS INJECTION AT THE BOUNDARIES. Abdel-Khalik, S.I.; Felde, D.K. (Univ. of Wisconsin, Madison). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

A volumetrically-heated pool with noncondensable gas injection at the boundaries is used to simulate the heat transfer processes taking place in molten core debris/concrete systems. Measurements of the upward, downward, and sideward heat transfer rates have been made over wide ranges of internal Rayleigh number ($6.0 \times 10^5 < Ra < 2.7 \times 10^9$), superficial gas velocity ($0 < V_g < 2.5$ cm/min), and pool aspect ratio ($1.32 < W/L < 5.16$). Pools with either an isothermal solid upper boundary or a free surface have been examined. Detailed measurements of the temperature distribution within the pool have also been made. The results indicate that the downward heat transfer rates are significantly increased while the sideward heat transfer rates are slightly affected by gas injection. The heat transfer rates at the pool boundaries are sensitive to the upper surface boundary condition.

- 109 (ANL--78-10, pp 361-368) BUBBLE-INDUCED HEAT TRANSFER IN A HEAT GENERATING LIQUID LAYER. Bergholz, R.F.; Bjorge, R. (Argonne National Lab., IL). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

The purpose of this investigation is to study the heat transfer behavior of a volumetrically heated liquid layer with gas bubbling from a horizontal, porous, lower boundary. The experimental data obtained are used to develop a correlation for the interfacial heat transfer coefficient for use in PAHR pool growth models. The experimental results also provide information on the upward heat loss due to gas and vapor flow through the liquid layer. Finally, the characteristics of bubble-induced heat transfer at a porous surface are compared with those found in experiments on nucleate boiling.

- 110 (ANL--78-10, pp 369-376) THERMO-FLUID MECHANICS OF VOLUME-HEATED BOILING POOLS. Greene, G.A.; Jones, D.C. Jr.; Schwarz, C.E. (Brookhaven National Lab., Upton, NY). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

A program to study the behavior of volume-heated boiling pools has been undertaken at BNL in order to provide meaningful estimates of heat transfer mechanisms in such systems. Emphasis has been placed on developing appropriate scaling parameters to allow results to be applied to prototypical, hypothetical accident conditions in an LMFBR. It has been determined that, within experimental accuracy, a boiling pool seems to exhibit behavior much like a single-phase natural convection system with a much extended laminar film regime to local Rayleigh numbers approximately 10^{12} .

- 111 (ANL--78-10, pp 389-397) SANDIA IN-CORE MOLTEN FUEL POOL EXPERIMENT PRELIMINARY RESULTS. Plein, H.G.; Carlson, G.A. (Sandia Labs., Albuquerque, NM). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov

1977).

The PAHR Molten Fuel Pool experimental program is concerned with the interaction of very hot fuel with reactor structural or sacrificial materials. The approach is to use fission heated fuel to obtain sustained, intrinsic heating which rather closely approximates the decay heated case. This experimental method also allows the fuel to interact with real structural and core retention materials at the actual temperatures of interest. The development of an advanced temperature diagnostic, ultrasonic thermometry, for the molten pool experiments; a materials compatibility study which defined the materials used for containment and insulation for the in-core experiments; and the results of the first three in-core molten pool experiments, two with UO_2 -only, and one containing both UO_2 and steel are presented. Plans for future molten pool studies are also discussed.

- 112 (ANL--78-10, pp 408-414) MOLTEN CORE DEBRIS INTERACTIONS WITH CORE CONTAINMENT MATERIALS. Swanson, D.G. (Ivan A. Getting Labs., El Segundo, CA); Zehms, E.H.; Ang, C.Y.; van Paassen, H.L.L. 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

A series of experiments have been performed to study the interactions expected in a hypothetical core meltdown accident between molten core debris and materials that might be used for core containment. The core debris would contain UO_2 and substantial quantities of stainless steel from the fuel cladding and support structures. Many sacrificial-bed concepts have been proposed for core containment in which materials are used that have been identified as having desirable thermal, chemical, and mechanical properties. In the experiments described here, the interactions between molten UO_2 and steel and various materials that might be employed have been investigated on a small scale. Molten UO_2 interactions with MgO brick, low-carbon steel plate, and stainless-steel plate have been studied, and interactions between molten stainless steel and MgO brick and low-carbon steel plate were also examined.

- 113 (ANL--78-10, pp 415-423) INTERACTIONS OF UO_2 WITH REACTOR MATERIALS. Fink, J.K.; Heiberger, J.J.; Leibowitz, L. (Argonne National Lab., IL). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

Two sets of in-progress small-scale interaction experiments are discussed. The interaction of UO_2 -graphite and stainless steel is being studied from 1894 to 2760 K for varied amounts of stainless steel (0 to 75%). Results of these experiments indicate that, below 2223 K, CO production is inhibited by the presence of stainless steel. Between 2223 K and 2760 K, the rate of CO evolution is significantly increased by the presence of stainless steel. The reaction rate increases with temperature. The interactions of Na_2O - Na_2CO_3 with UO_2 , mixtures of UO_2 and stainless steel, and ThO_2 at 1255 K are discussed. Na_2O - Na_2CO_3 interacts with UO_2 forming sodium uranates. Stainless steel particles dissolve in the Na_2O - Na_2CO_3 - UO_2 mixture to a limited extent. No reaction was observed between Na_2O - Na_2CO_3 and ThO_2 .

- 114 (ANL--78-10, pp 425-432) MEASUREMENT OF THE PENETRATION OF MOLTEN CORE MATERIALS INTO CONCRETE USING AN ACOUSTIC TECHNIQUE.

Sutherland, H.J. (Sandia Labs., Albuquerque, NM). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

In a recent article, Sutherland and Kent described a pulse-echo ultrasonic technique which measures the motion, as a function of time, of the interface of two dissimilar materials. They illustrated the technique by monitoring the motion of an ice/flame interface and a concrete/plasma jet interface, and they suggested that this technique could be used to monitor the motion of a molten pool/concrete interface. The technique, with some modifications, has now been used in this second configuration. These modifications and some preliminary results are reported.

- 115 (ANL--78-10, pp 433-440) SUSTAINED MOLTEN STEEL/CONCRETE INTERACTIONS TESTS. Powers, D.A. (Sandia Labs., Albuquerque, NM). 1977.

From 3. post-accident heat removal information exchange; Argonne, IL, USA (2 Nov 1977).

Four tests of the sustained interaction between high-temperature steel melts and two varieties of concrete are described. Inductive heating was used to keep the 200 Kg stainless steel melts liquid for periods of up to 20 min. A cover assembly over the melt-concrete system controlled emissions produced by the interactions so that quantification of phenomena produced in the tests could be attempted. Measurements of the gas composition, gas temperature, gas generation rate, aerosol emissions, upward heat-flux from the melt, and melt-temperature were made. Gases liberated from the concrete were found to be chemically reduced principally to carbon monoxide and hydrogen as they passed through the melt. The tests provided some evidence that stainless steel may also catalyze the hydrogenation of carbon monoxide to yield hydrocarbons such as methane, ethene, and ethane in the gas stream.

- 116 (ANL--78-17) ANALYTICAL STUDY OF THE FEASIBILITY OF IRRADIATING $^{233}\text{U}/^{232}\text{Th}$ METAL FUEL EXPERIMENTS IN EBR-II. Foltman, A.J.; Meneghetti, D. (Argonne National Lab., IL (USA)). Apr 1978. Contract W-31-109-ENG-38. 34p. AT.

Recent concerns about proliferation and diversion of plutonium have led to reconsideration of $^{233}\text{U}/^{232}\text{Th}$ fuel cycles. Although thorium fuels have been studied earlier, much of that work is incomplete; consequently, additional irradiation studies will be necessary. Experimental Breeder Reactor II (EBR-II) has the capability to perform many of these studies. Two broad types of irradiation experiments using $^{233}\text{U}/^{232}\text{Th}$ metal fuel are investigated: (1) full-core and/or blanket loadings and (2) individual experimental subassemblies loaded into current EBR-II cores. Calculations indicate that current EBR-II core dimensions give rise to ^{233}U enrichments as high as 48% and even higher for full-core loadings using $^{235}\text{U}/^{232}\text{Th}$ fuel. Because the uranium content in thorium is metallurgically restricted to 20 wt %, these enrichments are impractical. On the other hand, calculations show that it is feasible to irradiate up to six $^{233}\text{U}/^{232}\text{Th}$ metal-fueled subassemblies containing as much as 20 wt % uranium in thorium. Each experimental subassembly would be located in a row-7 "corner" of EBR-II's core. The local environment of the experiment would be tailored by surrounding the experiment with three subassemblies containing beryllium in row 8 and

three subassemblies containing half beryllium and half fueled driver elements in rows 6 and 7. Linear heat-generation rates of 40.4 to 53.1 kW/m are obtainable while heating in nearby rows increases less than 15%. Burnup should be about 2.9 to 3.6 at. % per year. Little degradation of the remaining core environment occurs, and the experiments would have only a modest impact on EBR-II's current irradiation program.

- 117 (ANL--78-23) PHYSICS OF REACTOR SAFETY. QUARTERLY REPORT, OCTOBER--DECEMBER 1977. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 32p. Dep. NTIS, PC A03/MF A01.

Work on the reactor safety program includes Monte Carlo analysis of safety-related critical assembly experiments by members of the Theoretical Fast Reactor Physics Group, and Planning of Safety-Related (ZPR) Planning and Experiments Group. Work on reactor core thermal-hydraulic code development performed in the Components Technology Division is also included.

- 118 (ANL--78-24) LOSS-OF-FLOW TEST L5 ON FFTF-TYPE IRRADIATED FUEL. Simms, R.; Gehl, S.M.; Lo, R.K.; Rothman, A.B. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 139p. Dep. NTIS, PC A07/MF A01.

Test L5 simulated a hypothetical loss-of-flow accident in an LMFBR using three (Pu, U) O_2 fuel elements of the FTR type. The test elements were irradiated before TREAT Test L5 in the General Electric Test Reactor to 8 at. % burnup at about 40 kW/m. The preirradiation in GETR caused a fuel-restructuring range characteristic of moderate-power structure relative to the FTR. The test transient was devised so that a power burst would be initiated at incipient cladding melting after the loss of flow. The test simulation corresponds to a scenario for FTR in which fuel in high-power-structure subassemblies slump, resulting in a power excursion. The remaining subassemblies are subjected to this power burst. Test L5 addressed the fuel-motion behavior of the subassemblies in this latter category. Data from test-vehicle sensors, hodoscope, and post-mortem examinations were used to construct the sequence of events within the test zone. From these observations, the fuel underwent a predominantly dispersive event just after reaching a peak power six times nominal at, or after, scram. The fuel motion was apparently driven by the release of entrained fission-product gases, since fuel vapor pressure was deliberately kept below significant levels for the transient. The test remains show a wide range of microstructural evolution, depending on the extent of heat deposition along the active fuel column. Extensive fuel swelling was also observed as a result of the lack of the cladding restraint. The results of the thermal-hydraulic calculations with the SAS3A code agreed qualitatively with the postmortem results with respect to the extent of the melting and the dispersal of cladding and fuel. However, the calculated times of certain events did not agree with the observed times.

- 119 (ANL--78-28) VISCOSITY OF MOLTEN ALUMINA. Blomquist, R.A.; Fink, J.K.; Leibowitz, L. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 18p. Dep. NTIS, PC A02/MF A01.

In the analysis of LMFBR core-containment and heat-removal problems associated with

- hypothetical core-disruptive accidents, viscosity data on molten ceramics are needed to help analyze the convective heat transfer and flow patterns within liquid pools. An oscillating cup viscometer has been used to measure the viscosity of molten alumina in the temperature range from 2400 to 2750°K. The data are represented by the equation: $\log \eta = 11448/T - 8.2734$ where the viscosity, η , is given in pascal seconds and the temperature, T , is in kelvins.
- 120 (ANL--78-30) TRITIUM METERS FOR USE IN LMFBF SODIUM COOLANT. Skladzien, S.B.; Renner, T.A.; Raue, D.J.; McPheeters, C.C. (Argonne National Lab., IL (USA)). Jun 1978. Contract W-31-109-ENG-38. 21p. AT.
- Tritium meters for continuously measuring tritium concentrations in the liquid metal coolant of fast breeder reactors have been designed, built, and tested. The tritium meters operate by diffusion of tritium from the sodium through a nickel membrane into a sweep-gas stream. The sweep gas flows to a gas-proportional-counting tube in which the tritium activity is measured. Tests on the Apparatus for Monitoring and Purifying sodium (AMPS) at ANL have shown excellent agreement between calculated and observed count rates under various conditions of sweep-gas flow rate and membrane temperature.
- 121 (ANL--78-31) STUDY OF THE TRANSIENT CORRECTION FACTOR USED IN TREAT REACTOR IN-PILE EXPERIMENTS. Graff, D.L. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 57p. Dep. NTIS, PC A04/MF A01.
- The transient correction factor (TCF) is used in TREAT reactor experiments to predict the power calibration coefficient (PC) for prototype LMFBF fuel pins at transient conditions without subjecting the pins to a reactor transient. The TCF is defined as the fissions in a monitor wire per MW of reactor power at transient conditions divided by the fissions in a monitor wire per MW of reactor power at low power conditions. The power calibration factor is the power in the target fuel per gram of ^{235}U per MW reactor power. The purpose of this study was to reduce the uncertainties concerning the transient correction factor by examining the nature and applicability of the TCF method. ANISN, a discrete-ordinates transport code, was used to simulate two TREAT reactor tests.
- 122 (ANL--78-34) TRANSIENT ENERGY TRANSFER BY CONDUCTION AND RADIATION FOR A SUDDEN CONTACT BETWEEN MOLTEN UO_2 AND SODIUM. Cho, D.H.; Chan, S.H. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 18p. Dep. NTIS, PC A02/MF A01.
- The transient energy transfer following a sudden contact between molten UO_2 and sodium has been investigated, taking into consideration both conduction and internal thermal radiation in UO_2 . Analytical expressions for the contact-interface temperature valid for small times are derived. Illustrative calculations indicate that on a time scale relevant to fuel-coolant interactions, internal radiation of molten UO_2 should have an insignificant effect on the contact-interface temperature between molten UO_2 and sodium. It thus appears that for the purpose of assessing the potential for an explosive fuel-coolant interaction, the contact-interface temperature may be adequately determined based on consideration of pure conduction.
- 123 (ANL--78-36) ROLE OF BUBBLE-SIZE EQUILIBRATION IN THE TRANSIENT BEHAVIOR OF FISSION GAS. Gruber, E.E. (Argonne National Lab., IL (USA)). Apr 1978. Contract W-31-109-ENG-38. 32p. Dep. NTIS, PC A03/MF A01.
- An explicit analysis of bubble-size equilibration by volume diffusion has been derived and incorporated into the FRAS2 code. This code was developed for mechanistic analysis of transient fission-gas behavior in LMFBF fuel, but had been limited by the approximation that the time required for bubble-size adjustment, following coalescence or changes in temperature or pressure, was negligible. The equilibration phenomenon is illustrated by several idealized examples. If the fuel temperature rises linearly with time, an isolated bubble will expand rapidly toward its equilibrium size only after the temperature reaches a critical value that depends on both bubble size and heating rate. Conversely, if the temperature is reduced uniformly, an isolated bubble will shrink only until diffusion becomes too slow for further size change. The final size that is "frozen in" may be significantly larger than the final equilibrium size. Finally, isothermal equilibration following coalescence of two equal bubbles can be described conveniently in terms of relaxation times, which depend on bubble size and temperature. However, a complete picture of the role of equilibration in transient fission-gas behavior can only be gained through an investigation of the bubble-size distribution and its evolution. Such an investigation is carried out with the FRAS2 code, in which the explicit equilibration model is implemented. Since nonequilibrium bubbles in a thermal transient are generally overpressured, the gas represents a source of energy that can conceivably disrupt the fuel. This "dispersive potential" is also calculated in the FRAS2 code.
- 124 (ANL--78-38) EXPERIMENTAL VALIDATION OF PTA-1 COMPUTER CODE FOR PRESSURE-TRANSIENT ANALYSIS INCLUDING THE EFFECT OF PIPE PLASTICITY. Youngdahl, C.K.; Kot, C.A.; Valentin, R.A. (Argonne National Lab., IL (USA)). Jul 1978. Contract W-31-109-ENG-38. 59p. AT.
- The PTA-1 code for computing pressure transients in piping networks includes a computational model to treat the significant effect of plastic deformation of the piping on pulse propagation. Stanford Research Institute has completed an experimental program on the response of piping systems to internal pressure pulses which plastically deform portions of the piping. This report makes extensive comparisons between PTA-1 computations and these experimental results. The excellent agreement obtained for both pressure histories and strain histories for all the experiments indicates that the PTA-1 computational model for pipe plasticity effects is accurate. The computation-experiment comparisons also permit a number of observations and conclusions to be made on other aspects of computational modeling of pressure transients, particularly with respect to pulse propagation around elbows.
- 125 (ANL--78-41) COMPARISONS OF SODIUM VOID AND DOPPLER REACTIVITIES IN LARGE OXIDE AND CARBIDE LMFBRS. Su, S.F. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 64p. Dep. NTIS, PC A04/MF A01.
- Sodium void and Doppler reactivities in two full scale (3000 MWth) LMFBRS are analyzed; one is fueled with UO_2 - PuO_2 and the other is fueled with UC - PuC . These two reactors are

analyzed for beginning of life as well as for beginning and end of equilibrium cycle conditions, and the variations of these two safety parameters with burnup are explained. A series of comparative analyses of these two and several hypothetical reactors are carried out to determine how differences in fuel type, sodium content, and heavy metal concentration between an oxide and a carbide reactor affect their sodium void and Doppler reactivities. The effect of the presence of control poison on sodium void reactivity is also addressed.

- 126 (ANL--78-50) F1 PHENOMENOLOGICAL TEST ON FUEL MOTION: FINAL REPORT. Palm, R.G.; Gehl, S.M.; Fink, C.L.; Dever, D.J.; Rothman, A.B. (Argonne National Lab., IL (USA)). May 1978. Contract W-31-109-ENG-38. 95p. Dep. NTIS, PC A05/MF A01.

TREAT F-series tests are being conducted to provide data on fuel motion in an LMFBR during a hypothetical loss-of-flow accident. Fuel and fuel-boundary conditions in an LMFBR subassembly following sodium voiding and dryout under loss-of-flow conditions are simulated in each F-series test. Simulation is achieved with a single fuel element surrounded by an annular nuclear-heated wall in a dry (no sodium) test capsule. The area inside the heated wall was selected to represent the area inside the perimeter of an LMFBR coolant channel. Test F1 was conducted with an irradiated fuel element to investigate the effect of fission gas on fuel motion at design power levels following cladding melting and drainage. The principal conclusion from Test F1 is that fission products retarded, but did not prevent, eventual fuel collapse. The collapse was retarded by a fuel/fission-product froth that prevented fuel collapse until the fission products separated from the partially molten fuel. The fuel motion observed in F1 represents a particular type of fuel (burnup of 2.35 at.%, power rating of 394 W/cm) transient heated at design power rating.

- 127 (ANL--78-51) NODAL METHOD FOR SOLVING TRANSIENT FEWGROUP NEUTRON DIFFUSION EQUATIONS. Shober, R.A. (Argonne National Lab., IL (USA)). Jun 1978. Contract W-31-109-ENG-38. 82p. AT.

A nodal method for multidimensional light water reactor (LWR) static and transient analysis is presented. This method efficiently solves one- or two-group diffusion equations using an analytic solution procedure. The report details significant improvements made to those aspects of the method previously reported in the literature. Eigenvalues and power distributions are presented for several static benchmark problems. Time-dependent results for a difficult two-dimensional BWR kinetics benchmark problem are presented. A reference solution for this benchmark problem is also presented. The results presented in the report are summarized, and suggestions are made as to appropriate ways to extend this work to multigroup fast breeder reactor analysis.

- 128 (ANL--78-56) PROSA-1: A PROBABILISTIC RESPONSE-SURFACE ANALYSIS CODE. Vaurio, J.K.; Mueller, C. (Argonne National Lab., IL (USA)). Jun 1978. Contract W-31-109-ENG-38. 157p. Dep. NTIS, PC A08/MF A01.

Techniques for probabilistic response-surface analysis have been developed to obtain the probability distributions of the consequences of postulated nuclear-reactor accidents. The uncertainties of the consequences are caused by the variability of the system and model input parameters used in

the accident analysis. Probability distributions are assigned to the input parameters, and parameter values are systematically chosen from these distributions. These input parameters are then used in deterministic consequence analyses performed by mechanistic accident-analysis codes. The results of these deterministic consequence analyses are used to generate the coefficients for analytical functions that approximate the consequences in terms of the selected input parameters. These approximating functions are used to generate the probability distributions of the consequences with random sampling being used to obtain values for the accident parameters from their distributions. A computer code PROSA has been developed for implementing the probabilistic response-surface technique. Special features of the code generate or treat sensitivities, statistical moments of the input and output variables, regionwise response surfaces, correlated input parameters, and conditional distributions. The code can also be used for calculating important distributions of the input parameters. The use of the code is illustrated in conjunction with the fast-running accident-analysis code SACO to provide probability studies of LMFBR hypothetical core-disruptive accidents. However, the methods and the programming are general and not limited to such applications.

- 129 (ANL--78-58) EBR-II FISSION-PRODUCT-SOURCE TEST NO. 1. EBR-II PROJECT. Strain, R.V.; Fogle, G.L.; Thresh, H.R.; Heinrich, R.R.; Fryer, R.M.; So, B.Y.C.; Lambert, J.D.B. (Argonne National Lab., IL (USA)). Aug 1978. Contract W-31-109-ENG-38. 42p. AT.

A fission-product source (FPS) was irradiated in EBR-II to provide data for calibrating the facility's fuel-element rupture detector (FERD), which is a delayed-neutron monitor, and germanium-lithium argon-scanning system (GLASS), a fission-gas-activity monitor. A metal alloy source, Ni-3.2 wt % U, provided quantitative recoil release of the fission-product nuclides. The source alloy, in tubular form, was irradiated as core-region segments of 18 capsules in the FPS subassembly. The irradiation showed that the response of the FERD was linear with increasing reactor power. The magnitude of the FERD signal was dependent on the local fission rate for the FPS and the flow path of the sodium carrying the delayed-neutron emitters. The relatively high fission-gas activity released by the FPS allowed accurate calibration of the GLASS under several modes of operation and provided data for verifying a gas-release model for the reactor.

- 130 (ANL--78-59) CREEP-FATIGUE LIFE PREDICTION FOR DIFFERENT HEATS OF TYPE 304 STAINLESS STEEL BY LINEAR-DAMAGE RULE, STRAIN-RANGE PARTITIONING METHOD, AND DAMAGE-RATE APPROACH. Maiya, P.S. (Argonne National Lab., IL (USA)). Jul 1978. Contract W-31-109-ENG-38. 22p. Dep. NTIS, PC A02/MF A01.

The creep-fatigue life results for five different heats of Type 304 stainless steel at 593°C (1100°F), generated under push-pull conditions in the axial strain-control mode, are presented. The life predictions for the various heats based on the linear-damage rule, strain-range partitioning method, and damage-rate approach are discussed. The appropriate material properties required for computation of fatigue life are also included.

- 131 (ANL--78-64) TRITIUM AND HYDROGEN TRANSPORT IN LMFBR SYSTEMS: EBR-II, CRBR, AND FFTF. Renner, T.A.; McPheeters, C.C.

(Argonne National Lab., IL (USA)). Sep 1978. Contract W-31-109-ENG-38. 37p. AT.

A tritium and hydrogen transport model has been employed to simulate concentration profiles, tritium losses to auxiliary containment systems, and cold trap burdens for EBR-II, CRBR, and FFTF. Experimental data from EBR-II were found to correlate well with calculated tritium and hydrogen profiles. A major change relative to previous transport models, namely, the inhibiting effect of oxide coatings on tritium permeation through reactor structural surfaces, has been incorporated into the current model. Tritium release rates to auxiliary systems where oxide barrier effects were included were predicted to be approximately two orders of magnitude lower than those for the reference case where structural surfaces were assumed to be totally oxide-free. Tritium releases during operation of large LMFBRs are expected to present essentially no hazard to the environment.

- 132 (ANL--78-69) COMBINED MOTION OF FUEL AND COOLANT DUE TO FUEL-COOLANT INTERACTIONS UNDER HIGH RAMP RATE REACTIVITY INSERTION. Chang, K.I.; Cho, D.H. (Argonne National Lab., IL (USA)). Jul 1978. Contract W-31-109-ENG-38. 52p. AT.

An analysis has been made of the combined motion of fuel and coolant due to fuel-coolant interactions following a massive fuel failure in a high-ramp overpower transient. The motion of fuel and coolant was described using a two-fluid model formulation in which the mixture of sodium liquid and vapor and of fission gas, on the one hand, and the fuel particles, on the other, were treated as two superimposed continua. The method of solution employed was a numerical procedure called the ACE method, a modified version of the IMF technique. Illustrative calculations were made for a fuel rupture length of 60 cm in the FFTF geometry. Three different modes of fragmentation were considered. First, upon pin failure, molten fuel is completely fragmented and uniformly distributed in the coolant (uniform fragmentation). Second, fragmentation is initiated at the bottom of the mixing zone and propagates upward (bottom-initiated progressive fragmentation). Third, fragmentation is initiated at the top of the mixing zone and propagates downward (top-initiated progressive fragmentation).

- 133 (ANL--78-70, pp 74-88) LMFBR AND GCFR SUPPORT WORK. Aug 1978.

Chemical Engineering Division research highlights, 1977.

Information is presented concerning reactor safety and physical property studies; sodium technology; and fuels and materials chemistry.

- 134 (ANL--78-70) CHEMICAL ENGINEERING DIVISION RESEARCH HIGHLIGHTS, 1977. (Argonne National Lab., IL (USA)). Aug 1978. Contract W-31-109-ENG-38. 185p. Dep. NTIS, PC A09/MF A01.

Separate abstracts are included for sections with information on lithium/metal sulfide batteries; electrochemical energy development; advanced fuel cell development; utilization of coal; magnetohydrodynamics technology; LMFBR and GCFR support work; fuel cycle studies; fusion reactor research; solar energy development; and basic energy science.

- 135 (ANL--78-73) FAILURE OF A HIGH-POWER PIN IN A SIMULATED $\$3/S$ TOP ACCIDENT: TEST E6 FINAL REPORT. Doerner, R.C.; Murphy, W.F.;

Stahl, D.; Stanford, G.S. (Argonne National Lab., IL (USA)). Aug 1978. Contract W-31-109-ENG-38. 147p. AT.

This report describes the Fuel Dynamics Test E6 and analyzes the test data. A cluster of six fresh FTR-type fuel pins surrounding a previously irradiated pin was tested to failure in a simulated $\$3/S$ FFTF accident. It is shown that: (1) the central 34.3-cm-long test pin is thermodynamically equivalent to a one-third scale model of the 91.4-cm-long FFTF pins; (2) failure occurred 70 ms earlier than predicted in the PSAR; (3) postfailure fuel motion was principally upward in direction and from the center of the cluster, with no more than 10 g of total fuel loss; (4) the outlet blockage was formed in stages; (5) the irradiated fuel is more mobile than fresh fuel; (6) there were no significant fuel-coolant interactions; and (7) the lower one-third of the test pin suffered only minor cladding damage and no inlet blockage was formed. It is concluded that the mechanisms that led to initial failure can be reliably applied to the FTR. Characterization of the initial fuel release in amount, location, structure (solid, liquid or foam), and rate were reasonably prototypic. Postfailure coolant dynamics, however, were not very prototypic. Although the amount of fuel swept out is probably realistic, the formation of an outlet blockage and its impact on adjacent channel hydraulics is clearly nonprototypic.

- 136 (ANL--78-74) METALLOGRAPHIC AND FRACTOGRAPHIC OBSERVATIONS OF POSTTEST CREEP-FATIGUE SPECIMENS OF WELD-DEPOSITED TYPE 308 CRE STAINLESS STEEL. Williams, M.W. (Argonne National Lab., IL (USA)). Aug 1978. Contract W-31-109-ENG-38. 19p. AT.

Type 308 CRE stainless steel weld specimens were subjected to metallographic and fractographic analysis after failure in elevated-temperature (593°C) creep-fatigue tests. The failure mode for specimens tested under continuous-cycle fatigue conditions was predominantly transgranular. When the test cycle was modified to include a hold time at the maximum tensile strain, the failure mode became predominantly interphase. Sigma phase was observed within the delta-ferrite regions in the weld. However, the presence of sigma phase did not appear to affect the failure mode.

- 137 (ANL--78-84) ULTRASONIC SCANNER FOR STAINLESS STEEL WELD INSPECTIONS. Kupperman, D.S.; Reimann, K.J. (Argonne National Lab., IL (USA)). Sep 1978. Contract W-31-109-ENG-38. 34p. AT.

The large grain size and anisotropic nature of stainless steel weld metal make conventional ultrasonic testing very difficult. A technique is evaluated for minimizing the coherent ultrasonic noise in stainless steel weld metal. The method involves digitizing conventional "A-scan" traces and averaging them with a minicomputer. Results are presented for an ultrasonic scanner which interrogates a small volume of the weld metal while averaging the coherent ultrasonic noise.

- 138 (ANL--78-86) SECOND MECHANIZED INSERVICE INSPECTION OF EBR-II STEAM GENERATOR. Longua, K.J.; Whitham, G.K.; Allen, C.C.; Larson, H.A. (Argonne National Lab., IL (USA)). Nov 1978. Contract W-31-109-ENG-38. 18p. AT.

The second mechanized ultrasonic inspection of evaporator EV-702, part of the EBR-II steam-generator system, was completed in March 1978.

The 2 1/4 Cr-1 Mo duplex tubes were inspected from the bore (water) side for flaws and changes in wall thickness and for evaluating the condition of the braze between the tubes. An addition to this second inspection was the use of an ultrasonic probe to search for circumferential flaws. The inspection equipment was thoroughly checked and routinely calibrated, using a reference standard containing electrical-discharge-machined notches. Notches as small as 1.6 mm long by 0.25 mm deep were readily detected.

- 139 (ANL--78-87) DISPOSAL OF RADIOACTIVE SODIUM WASTE. McPheeters, C.C.; Wolson, R.D. (Argonne National Lab., IL (USA)). Sep 1978. Contract W-31-109-ENG-38. 25p. AT.

Operation of liquid-metal-cooled fast breeder reactors (LMFBRs) will result in production of various quantities of radioactive sodium waste. Two methods have been developed and tested on a small scale for converting this sodium waste to inert compounds suitable for disposal. The first method is direct oxidation of the sodium after dispersal in a silica matrix. The sodium is mixed with silica and oxidized in a rotary drum reactor. The product is suitable for making glass when other stabilizing compounds are added. The second method is reaction of elemental sodium with molten NaOH at 450°C and subsequent injection of steam into the melt to convert the reaction products (Na_2O and NaH) to additional NaOH. The reactions are smooth and easily controlled with little danger of runaway reactions. The end product is molten NaOH which can be cast into drums for further treatment or disposal. The advantages of these two methods over more conventional aqueous processes are the elimination of aqueous wastes and the elimination or minimization of gaseous effluents. 9 figures, 3 tables.

- 140 (ANL--78-89) FINAL SAFETY ANALYSIS ADDENDUM TO HAZARD SUMMARY REPORT, EXPERIMENTAL BREEDER REACTOR NO. II (EBR-II): THE EBR-II COVER-GAS CLEANUP SYSTEM. Fryer, R.M.; Monson, L.R.; Price, C.C.; Hooker, D.W. (Argonne National Lab., IL (USA)). Apr 1979. Contract W-31-109-ENG-38. 91p. AT.

This report evaluates abnormal and accident conditions postulated for the EBR-II cover-gas cleanup system (CGCS). Major considerations include loss of CGCS function with a high level of cover-gas activity, loss of the liquid-nitrogen coolant required for removing fission products from the cover gas, contamination of the cover gas from sources other than the reactor, and loss of system pressure boundary. Calculated exposures resulting from the maximum hypothetical accident (MHA) are less than 2% of the 25-Rem limit stipulated in U.S. Regulation 10 CFR 100; i.e., a person standing at any point on an exclusion boundary (area radius of 600 m) for 2 h following onset of the postulated release would receive less than 0.45 Rem whole-body dose. The on-site whole-body dose (10 m from the source) would be less than 16 Rem.

- 141 (ANL--78-92) LIQUID-LIQUID CONTACT IN VAPOR EXPLOSION. Segev, A. (Argonne National Lab., IL (USA)). Aug 1978. Contract W-31-109-ENG-38. 188p. Dep. NTIS, PC A09/MF A01.

The contact of two liquid materials, one of which is at a temperature substantially above the boiling point of the other, can lead to fast energy conversion and a subsequent shock wave. This phenomenon is called a vapor explosion. One method of producing intimate,

liquid-liquid contact (which is known to be a necessary condition for vapor explosion) is a shock tube configuration. Such experiments in which water was impacted upon molten aluminum showed that very high pressures, even larger than the thermodynamic critical pressure, could occur. The mechanism by which such sharp pressure pulses are generated is not yet clear. The report describes experiments in which cold liquids (Freon-11, Freon-22, water, or butanol) were impacted upon various hot materials (mineral oil, silicone oil, water, mercury, molten Wood's metal or molten salt mixture).

- 142 (ANL--78-97) FX2-TH: A TWO-DIMENSIONAL NUCLEAR REACTOR KINETICS CODE WITH THERMAL-HYDRAULIC FEEDBACK. Shober, R.A.; Daly, T.A.; Ferguson, D.R. (Argonne National Lab., IL (USA)). Oct 1978. Contract W-31-109-ENG-38. 380p. Dep. NTIS, PC A17/MF A01.

FX2-TH is a two-dimensional, time-dependent nuclear reactor kinetics program with thermal and hydraulic feedback. The neutronics model used is multigroup neutron diffusion theory. The following geometry options are available: x, r, x-y, r-z, theta-r, and triangular. FX2-TH contains two basic thermal and hydraulic models: a simple adiabatic fuel temperature calculation, and a more detailed model consisting of an explicit representation of a fuel pin, gap, clad, and coolant. FX2-TH allows feedback effects from both fuel temperature (Doppler) and coolant temperature (density) changes. FX2-TH will calculate a consistent set of steady state conditions by iterating between the neutronics and thermal-hydraulics until convergence is reached. The time-dependent calculation is performed by the use of the improved quasistatic method. A disk editing capability is available. FX2-TH is operational on IBM system 360 or 370 computers and on the CDC 7600.

- 143 (ANL--78-99) RESULTS OF UO_2 DIAGNOSTIC TESTS FOR RUN BEYOND CLADDING BREACH IN EBR-II. Strain, R.V.; Lambert, J.D.B. (Argonne National Lab., IL (USA)). Feb 1979. Contract W-31-109-ENG-38. 36p. AT.

Predefected UO_2 fuel elements were irradiated in EBR-II to determine the response of the fission-product-monitoring systems. In particular, the tests were performed to indicated whether the amount of oxide fuel exposed to the primary sodium could be estimated by monitoring, with the fuel-element-rupture detector (FERD), delayed neutrons (DN's) emitted from bromine and iodine isotopes in the sodium. Elements with defect areas of 0.005 and 0.20 cm^2 were irradiated consecutively in April and June 1977. The smaller defect produced no measurable DN signal, and the larger defect produced a DN signal of approx. 55 counts per second (cps) while irradiated at 11.3 kW/ft (3.44 kW/m) and 70 cps at 14.3 kW/ft (4.35 kW/m). The signal of 55 cps was about 20 times the signal that would be expected if the release from the defect had been solely by recoil from the exposed fuel. Postirradiation examination of the elements showed that an unexpectedly large amount of UO_2 /sodium chemical reaction occurred in the element with the 0.20- cm^2 defect.

- 144 (ANL--78-100) THEORY AND APPLICATION OF A QUASI-EULERIAN FLUID ELEMENT FOR THE STRAW CODE. Kennedy, J.M.; Belytschko, T.B. (Argonne National Lab., IL (USA)). Oct 1978. Contract W-31-109-ENG-38. 51p. Dep. NTIS, PC A04/MF A01.

Two-dimensional finite-element models for the treatment of the nonlinear, transient

response of fluids and structures are described. The fluid description is quasi-Eulerian, so that the mesh can move independently of the material, and it includes a new finite-element upwinding scheme. The structural description is based on a corotational formulation in which the coordinate system is embedded in the elements, which is applicable to arbitrarily large rotations. The interface between the fluid and structure permits relative sliding, but because of the description of the quasi-Eulerian fluid, the nodes of the fluid and structure can be allowed to remain contiguous. Modeling procedures for treating the various aspects of subassemblies, such as the narrow fluid channels, the fuel bundles which are immersed in the coolant, and the axial flow are developed. Calculations are made for a symmetric 7-subassembly cluster and compared to experimental results. In addition, the application to a 19-subassembly cluster is described.

- 145 (ANL--78-102) FLUCTUATION ANALYSIS OF FAST REACTOR SAFETY EXPERIMENTS IN TREAT. Doerner, R.C.; Meek, C.C.; Hurt, R.F.; Pekarsky, M.I. (Argonne National Lab., IL (USA)). Nov 1978. Contract W-31-109-ENG-38. 71p. AT.

Statistical fluctuations of measured signals about their mean are related to physical processes in fuel-failure experiments. Signal variance, correlation, and spectral density are shown to be sensitive measures of instrument response characteristics, of flow-blockage formation, and of boiling phenomena. This sensitivity is demonstrated by a series of examples that use test data from the E6, E7, and L5 experiments. A mathematical model of the Mark-II loop is developed to predict both the mean and the fluctuation behavior of measured test parameters. The analysis is extended to include signal forecasting by the ARIMA time-series model. Techniques that are used to identify the model and to estimate the model parameters are discussed in detail. It is shown that departure of real-time data from the on-line forecasts is a powerful tool for the rapid detection of off-normal conditions. A description of the experiments and the data-reduction process is given.

- 146 (ANL--78-103) ANALYSIS OF NONLINEAR FLUID STRUCTURE INTERACTION TRANSIENT IN FAST REACTORS. Wang, C.Y. (Argonne National Lab., IL (USA)). Nov 1978. Contract W-31-109-ENG-38. 60p. Dep. NTIS, PC A04/MF A01.

A generalized Eulerian method is described for analyzing the fluid transients and the structural response in nuclear reactors under the postulated accident conditions. The phenomena considered are the wave propagation, slug impact, sodium spillage, bubble migration, and the fluid-structure interaction. The basic equations and numerical formulation are presented in detail. Sample calculations are given to illustrate the analysis. It is shown from the results that the implicit, iterative method used is unconditionally stable and is especially suitable for problems involving large material distortions.

- 147 (ANL--78-104) INVESTIGATIONS OF MATERIALS COMPATIBILITY RELEVANT TO THE EBR-II SYSTEM, FY 1978. Longue, K.J.; Shields, J.A.; Ruther, W.E.; Olson, W.H. (Argonne National Lab., IL (USA)). Jan 1979. Contract W-31-109-ENG-48. 23p. AT.

Thermocouple assemblies used to monitor sodium piping failed in FY 1977. The

examination in FY 1978 showed that these corrosion failures of the thermocouples have not caused any degradation of the sodium piping. Evaporator EV-702 was ultrasonically inspected for the second annual inspection. The in-service inspection of the steam system and the secondary sodium system did not reveal any evidence of material degradation or defects in any pipe welds. An EBR-II periscope used to examine reactor internals became inoperative owing to deposition of sodium aerosol on its faceplate. A method was devised for preventing such deposition. Various illumination sources for the periscope were evaluated, and a GE sealed-beam lamp was recommended. Notch tensile tests did not detect any difference in notch sensitivity between 12% cold-worked AISI 316 and annealed AISI 304 stainless steels after irradiation.

- 148 (ANL--78-105) ICEPEL ANALYSIS OF AND COMPARISON WITH SIMPLE ELASTIC--PLASTIC PIPING EXPERIMENTS. A-Moneim, M.T. (Argonne National Lab., IL (USA)). Dec 1978. Contract W-31-109-ENG-38. 60p. Applied Technology.

The results of simple elastic--plastic piping experiments for straight pipes and single-elbow loop systems are interpreted and evaluated. The experiments are also analyzed by the ICEPEL piping code, and the analytical results are compared against the experimental data. Good agreement is obtained in predicting both pressure and circumferential strain histories along the pipe systems. Inside the elbow, however, the analysis showed radial pressure variation while the experiments showed no radial pressure variation. Consequently, the ICEPEL elbow model was modified to resolve this discrepancy. The new model agreed with the experiments in showing no radial pressure variation.

- 149 (ANL--79-7) DEVELOPMENT OF A MK-II LOOP TO SIMULATE REACTOR HYDRAULIC CONDITIONS. Page, R.J.; Robinson, L.E. (Argonne National Lab., IL (USA)). Jan 1979. Contract W-31-109-ENG-38. 35p. AT.

The Mk-II Integral Loop was modified to provide an in-pile experimental apparatus that would simulate the subassembly coolant flow rate and inlet pressure head of the Fast Test Reactor (FTR). There were two main design changes. First, the safety dump tanks were removed from the Mk-II loop and replaced by a second annular linear induction pump (ALIP). Second, a flow-restricting orifice was sized so that the hydraulic requirements of prototypical test-section coolant velocity (670 cm/s) and pressure head (0.74 MPa) would be achieved. In support of this effort, out-of-pile experiments were conducted to determine the frictional pressure drop across a seven-pin fuel bundle; determine the pumping characteristics of the dual ALIP system; determine the calibration characteristics of several sharp-edge orifices; and improve the test-section outlet flowmeter sensitivity. The resulting redesigned loop was used for the in-pile TREAT transient over-power Test H6, which investigated fuel sweepout and coolability following fuel-pin failure under hydraulic conditions typical of the FTR.

- 150 (ANL--79-13) SUMMARY OF TREAT EXPERIMENTS ON OXIDE CORE-DISRUPTIVE ACCIDENTS. Dickerman, C.E.; Rothman, A.B.; Klickman, A.E.; Spencer, B.W.; DeVolpi, A. (Argonne National Lab., IL (USA)). Feb 1979. Contract W-31-109-ENG-38. 59p. Dep. NTIS, PC A04/MF A01.

A program of transient in-reactor experiments is being conducted by Argonne

National Laboratory in the Transient Reactor Test (TREAT) facility to guide and support analyses of hypothetical core-disruptive accidents (HCDA) in liquid-metal fast breeder reactors (LMFBR). Test results provide data needed to establish the response of LMFBR cores to hypothetical accidents producing fuel failure, coolant boiling, and the movement of coolant, molten fuel, and molten cladding. These data include margins to fuel failure, the modes of failure and movements, and evidence for identification of the mechanisms which determine the failure and movements. A key element in the program is the fast-neutron hodoscope, which detects fuel movement as a function of time during experiments.

- 151 (ANL--79-14) WASTE PRODUCTION AND MANAGEMENT AT EBR-II. Witbeck, L.C.; Fryer, R.M. (Argonne National Lab., IL (USA)). Apr 1979. Contract W-31-109-ENG-38. 43p. AT.

This report surveys waste production rates and disposal practices at EBR-II, a small-scale liquid-metal fast breeder reactor, for the past decade. Normal airborne wastes have been reduced by switching to low-sulfur fuel oil in auxiliary boilers and by converting to use of reactor steam for part of the plant space heating. Atmospheric releases of gaseous radioactive wastes initially increased during the implementation of programs for testing reactor fuels to and beyond cladding breach; but the effluent rates have now been reduced to far below permissible limits by use of a cryogenic distillation column. EBR-II's small production of liquid radioactive waste is handled by evaporation, followed by disposal of the evaporator sludge as solid waste. Liquid-waste releases meet all applicable state and federal standards. Solid wastes are sent to two storage/disposal areas - one of which is reserved primarily for storage of materials containing sodium, pending development of a facility for disposal of the sodium. In overall waste emission, EBR-II compares very favorably with commercial light-water reactors.

- 152 (ANL--79-26) FISSION-PRODUCT RELEASES TO THE PRIMARY SYSTEM OF EBR-II FROM APRIL 1975 TO MARCH 1977. So, B.Y.C.; Lambert, J.D.B.; Kirn, F.S.; Armstrong, J.R.; Ebersole, E.R.; Laug, M.T. (Argonne National Lab., IL (USA)). May 1979. Contract W-31-109-ENG-38. 59p. AT.

This report describes the 14 releases of fission products that occurred in EBR-II from April 1975 to March 1977. Each release was readily detected, and all but one (in a driver-fuel subassembly) was identified with a particular subassembly. Xenon tagging was the primary method of identification, although other methods were used where appropriate. Methods of monitoring and identifying fission-product sources are discussed, and each release and identification is described. Effects of breached elements on plant availability were minimal in this period. From all evidence, cladding breaching on elements in EBR-II continues to be a benign process.

- 153 (ANL--79-27) FISSION-PRODUCT RELEASES TO THE PRIMARY SYSTEM OF EBR-II FROM APRIL 1977 TO MAY 1978. So, B.Y.C.; Gross, K.C.; Lambert, J.D.B.; Kirn, F.S.; Ebersole, E.R.; Laug, M.T. (Argonne National Lab., IL (USA)). Jul 1979. Contract W-31-109-ENG-38. 27p. AT.

Suspected fission-product releases from 18 subassemblies between April 1977 and May 1978 are described. Postirradiation examinations on 15 of the suspect subassemblies confirmed that

all contained one or more breached elements. Except for two untagged subassemblies, xenon tagging was the primary method of identification, although other methods were used where appropriate. Methods to monitor and identify fission-product sources are discussed. Problems encountered with multiple-element breaches are described. Overall, the effects of breached elements on plant availability were minimal during this reporting period. From all evidence, cladding breaching on elements in EBR-II continues to be a benign process.

- 154 (ANL--79-57) DEVELOPMENT OF A METHODOLOGY FOR ANALYSIS OF DELAYED-NEUTRON SIGNALS. Gross, K.C.; Strain, R.V.; Fryer, R.M. (Argonne National Lab., IL (USA)). Feb 1980. Contract W-31-109-ENG-38. 84p. NTIS, PC A05/MF A01.

Experimental and analytical techniques have been developed for analysis and characterization of delayed-neutron (DN) signals that can provide diagnostic information to augment data from cover-gas analyses in the detection and identification of breached elements in an LMFBR. Eleven flow-reduction tests have been run in EBR-II to provide base data support for predicting DN signal characteristics during exposed-fuel operation. Results from the tests demonstrate the feasibility and practicability of response-analysis techniques for determining (a) the transit time, $T_{\text{sub tr}}$, for DN emitters traveling from the core to the detector and (b) the isotropic holdup time, $T_{\text{sub h}}$, of DN precursors in the fuel element.

- 155 (ANL--79-63) MECHANICAL PROPERTIES TEST PROGRAM ON STRUCTURAL MATERIALS IN A SODIUM ENVIRONMENT. Natesan, K.; Chopra, D.K.; Kassner, T.F. (Argonne National Lab., IL (USA)). Oct 1979. Contract W-31-109-ENG-38. 20p. Dep. NTIS, PC A02/MF A01.

This document describes in detail the ongoing and planned US Test program on the mechanical properties of sodium-exposed Type 316 austenitic stainless and Fe-2 1/4 Cr-1 Mo ferritic steels. The test program is based on the Development Requirement Specifications (DRS) established by the DOE/Clinch River Breeder Reactor Project (CRBRP) Program Office, the general need for the development of LMFBR structural-design criteria established by the Nuclear Systems Materials Handbook, and the need for a fundamental understanding of materials behavior in a sodium environment, which is generic to LMFBR systems. The planned test program is an extension of work based on current knowledge of sodium chemistry and the influence of sodium purity on the mechanical properties of structural materials.

- 156 (ANL--79-70) NUMERICAL METHOD FOR SOLUTION OF TRANSIENT, HOMOGENEOUS, EQUILIBRIUM, TWO-PHASE FLOWS IN ONE SPACE DIMENSION. Shin, Y.W.; Wiedermann, A.H. (Argonne National Lab., IL (USA)). Oct 1979. Contract W-31-109-ENG-38. 28p. Dep. NTIS, PC A03/MF A01.

A solution method is presented for transient, homogeneous, equilibrium, two-phase flows of a single-component fluid in one space dimension. The method combines a direct finite-difference procedure and the method of characteristics. The finite-difference procedure solves the interior points of the computing domain; the boundary information is provided by a separate procedure based on the characteristics theory. The solution procedure for boundary points requires information in addition to the physical boundary conditions.

This additional information is obtained by a new procedure involving integration of characteristics in the hodograph plane. Sample problems involving various combinations of basic boundary types are calculated for two-phase water/steam mixtures and single-phase nitrogen gas, and compared with independent method-of-characteristics solutions using very fine characteristic mesh. In all cases, excellent agreement is demonstrated.

- 157 (ANL--79-72) ALICE: A HYBRID LAGRANGIAN-EULERIAN CODE FOR CALCULATING FLUID-STRUCTURE INTERACTION TRANSIENTS IN FAST-REACTOR CONTAINMENT. Chu, H.Y. (Argonne National Lab., IL (USA)). Aug 1979. Contract W-31-109-ENG-38. 59p. Dep. NTIS, PC A04/MF A01.

This report describes a reactor-containment code, ALICE, which uses an arbitrary Lagrangian-Eulerian method to describe the coolant motion, together with a Lagrangian method to analyze the response of the containment vessel and other solid media inside a reactor containment. The finite-difference formulation used to approximate the governing equations for the motion of the coolant can be solved in either an explicit or an implicit scheme; the finite-element formulation used to approximate the governing equations for the containment vessel and other solid media can be performed only in the explicit scheme. Thus, the ALICE code can perform two types of coupling calculations for the fluid and structure (implicit-explicit and explicit-explicit). The code is generalized so that it can apply to problems either in a two-dimensional Cartesian or in a two-dimensional cylindrical-coordinate system.

- 158 (ANL--79-73) MULTICHANNEL CLAD-RELOCATION MODEL FOR FAST-REACTOR LOSS-OF-FLOW ACCIDENTS. Ishii, M.; Chen, W.L.; Grohmes, M.A. (Argonne National Lab., IL (USA)). Aug 1979. Contract W-31-109-ENG-38. 62p. Dep. NTIS, PC A04/MF A01.

During an unprotected undercooling accident in a liquid-metal fast breeder reactor, the motion and relocation of the molten cladding can be important because of its potentially significant effect on reactivity, blockage formation, and subsequent fuel motion. The present study analyzes the clad relocation problem based on a multichannel film-flow model. The important aspects considered in the analysis are the non-uniform transverse clad-melting pattern and diversions of sodium-vapor flow within a subassembly. It has been shown that the motion of molten clad and subsequent blockage formations can be significantly influenced by this interconnected channel effect. Several sample calculations have been made in order to demonstrate these points.

- 159 (ANL--79-86) LASER MICROSAMPLING METHOD FOR DETERMINATION OF RETAINED FISSION GAS IN IRRADIATED NUCLEAR FUELS. Graczyk, D.G.; Bandyopadhyay, G.; Gehl, S.M.; Hughes, J.P.; Goodspeed, H.T. (Argonne National Lab., IL (USA)). Oct 1979. Contract W-31-109-ENG-38. 41p. Dep. NTIS, PC A03/MF A01.

A small ruby laser adapted to fire through a microscope is used to release fission gases from specific sites on a plane surface of an irradiated fuel specimen. Interaction of the focused laser pulse with the specimen surface results in a conical crater from which sampled material has been vaporized; the crater is surrounded by a heat-affected zone in which intergranular fracture and grain separation allow release of grain-boundary gases. Procedures for measuring the amount of krypton-

85 released by laser heating and the volume of material from which the release occurred are presented. The data obtained may be used to obtain local krypton fission-gas concentrations and the intragranular/intergranular distribution.

- 160 (ANL--79-89) QUASI-EULERIAN METHOD FOR ANALYZING SLUG IMPACT AND COOLANT SPILLAGE IN A FAST-REACTOR ACCIDENT. Chu, H.Y. (Argonne National Lab., IL (USA)). Dec 1979. Contract W-31-109-ENG-38. 31p. Dep. NTIS, PC A03/MF A01.

This report describes a quasi-Eulerian method which has been incorporated into the ICECO code to study slug impact and coolant spillage problems in a fast-reactor accident. The quasi-Eulerian cells used in this method are located on the tops of the regular cells. The axial size of the quasi-Eulerian cells varies according to the gap generated at the reactor head-wall junction. Penetration holes on the cover head are modeled on the top center of the quasi-Eulerian cells. Fluid variables in these quasi-Eulerian cells also satisfy all the conservation equations. Since the boundary pressures above the quasi-Eulerian cells are determined by the movement of the moving grid, the velocity of the cover head is also included in the pressure iteration. Several examples are given to compare the results obtained by this quasi-Eulerian method with the existing experimental excursion data, as well as with the analytical and the other code solutions.

- 161 (ANL--79-91) LIMBO COMPUTER CODE FOR ANALYZING COOLANT-VOIDING DYNAMICS IN LMFBR SAFETY TESTS. Bordner, G.L. (Argonne National Lab., IL (USA)). Oct 1979. Contract W-31-109-ENG-38. 34p. Dep. NTIS, PC A03/MF A01.

The LIMBO (liquid metal boiling) code for the analysis of two-phase flow phenomena in an LMFBR reactor coolant channel is presented. The code uses a nonequilibrium, annular, two-phase flow model, which allows for slip between the phases. Furthermore, the model is intended to be valid for both quasi-steady boiling and rapid coolant voiding of the channel. The code was developed primarily for the prediction of, and the posttest analysis of, coolant-voiding behavior in the SLSF P-series in-pile safety test experiments. The program was conceived to be simple, efficient, and easy to use. It is particularly suited for parametric studies requiring many computer runs and for the evaluation of the effects of model or correlation changes that require modification of the computer program. The LIMBO code, of course, lacks the sophistication and model detail of the reactor safety codes, such as SAS, and is therefore intended to compliment these safety codes.

- 162 (ANL--79-104) FINITE-ELEMENT METHOD FOR ABOVE-CORE STRUCTURES. Kennedy, J.M.; Belytschko, T.B. (Argonne National Lab., IL (USA)). Dec 1979. Contract W-31-109-ENG-38. 50p. Dep. NTIS, PC A03/MF A01.

Three-dimensional finite-element models for the treatment of the nonlinear, transient response of a fast breeder reactor's above-core structures are described. For purposes of treating arbitrarily large rotations, node orientations are described by unit vectors and the deformable elements are treated by a corotational formulation in which the coordinate system is embedded in the elements. Deformable elements may be connected either to nodes directly or through rigid bodies. The time integration is carried out by the Newmark

β method. These features have been incorporated to form the finite-element program SAFE/RAS (Safety Analysis by Finite Elements/Reactor Analysis and Safety Division). Computations are presented for semianalytical comparisons, simple scoping studies, and Stanford Research Institute (SRI) test comparisons.

- 163 (ANL--80-31) ACCURATE NUMERICAL SOLUTIONS FOR ELASTIC-PLASTIC MODELS. Schreyer, H.L.; Kulak, R.F.; Kramer, J.M. (Argonne National Lab., IL (USA)). Mar 1980. Contract W-31-109-ENG-38. 26p. NTIS, PC A03/MF A01.

The accuracy of two integration algorithms is studied for the common engineering condition of a von Mises, isotropic hardening model under plane stress. Errors in stress predictions for given total strain increments are expressed with contour plots of two parameters: an angle in the π plane and the difference between the exact and computed yield-surface radii. The two methods are the tangent-predictor/radial-return approach and the elastic-predictor/radial-corrector algorithm originally developed by Mendelson. The accuracy of a combined tangent-predictor/radial-corrector algorithm is also investigated.

- 164 (ANL--7542(Add.C)) CATALOG OF ROD-DROP DATA FROM EBR-II RUNS 67-87. Foltman, A.J. (Argonne National Lab., IL (USA)). Sep 1977. Contract W-31-109-ENG-38. 40p. AT.

Kinetics studies using the rod-drop procedure have been conducted in EBR-II for runs 67 to 87. Most of the studies were routine measurements of the prompt power coefficient. However, additional tests aided in determining limitations and errors of the rod-drop method that in turn helped lead to a full-power calibration procedure. The number of drops necessary for acceptable accuracy was studied as well; five drops per test was suggested for EBR-II. An additional test permitted us to determine that at least 24 hours are needed to allow the core to reach thermal and mechanical equilibrium in terms of reactivity effects.

- 165 (ANL--7721) ARC SYSTEM FUEL CYCLE ANALYSIS CAPABILITY, REBUS-2. Hosteny, R.P. (Argonne National Lab., IL (USA)). Oct 1978. Contract W-31-109-ENG-38. 409p. Dep. NTIS, PC A18/MF A01.

A detailed description is given of the ARC System fuel cycle modules FCI001, FCC001, FCC002, and FCC003 which form the fuel cycle analysis modules of the ARC System. These modules, in conjunction with certain other modules of the ARC System previously described in documents of this series, form the fuel cycle analysis system called REBUS-2. The physical model upon which the REBUS-2 fuel cycle modules are based and the calculational approach used in solving this model are discussed in detail. The REBUS-2 system either solves for the infinite time (i.e., equilibrium) operating conditions of a fuel recycle system under fixed fuel management conditions, or solves for the operating conditions during each of a series of explicitly specified (i.e., nonequilibrium) sequence of burn cycles. The code has the capability to adjust the fuel enrichment, the burn time, and the control poison requirements in order to satisfy user specified constraints on criticality, discharge fuel burnup, or to give the desired multiplication constant at some specified time during the reactor operation.

- 166 (ANL/A--78-3) STUDY OF BASIC APPROACHES TO THE DESIGN OF THE PRIMARY SYSTEM OF LOOP-TYPE LMFBRs. Amorosi, A.; Ahmed, H.; Hutter, E. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 250p. Dep. NTIS, PC A11/MF A01.

The report describes the results of a study conducted by an informal "Primary System Study Group" formed at ANL to review from a systems point of view the many possible approaches available to the design of the primary heat transport system for large-size (roughly, 1200 MWe), loop-type LMFBRs. The study is based on examination of the various design options which have been used or proposed to date in the principal loop-type projects and design studies in this country and abroad. The objective of the report is to provide an overview of the approaches which have been considered over the years, to identify and discuss some of their advantages and disadvantages, and, where possible, to indicate which approaches appear to warrant further consideration. The approaches discussed in the report differ principally in: the use of different sizes of reactor vessel--small, with external piping downcomers, or large, with internal downcomers; use of different pump locations--hot leg or cold leg; and, use of different methods for accommodation of system thermal expansion--pipe and elbow loops, special expansion accommodators, free-end pipes, free component movement, and others. The important characteristics, from the systems standpoint, and some particularly pertinent design considerations of the major primary system components are also examined, with emphasis on reactor vessel, pump, and intermediate heat exchanger.

- 167 (ANL-AFP--2) ADVANCED FUELS PROGRAM. QUARTERLY PROGRESS REPORT FOR THE PERIOD, JULY--SEPTEMBER 1974. Kittel, J.H. (comp.). (Argonne National Lab., IL (USA)). 1974. Contract W-31-109-ENG-38. 136p. AT.

Progress is summarized in the following LMFBR mixed carbide and mixed nitride fuel development tasks: (1) fuel system design studies; (2) fuel element modelling; (3) steady state irradiations; (4) off-normal irradiations; (5) fuel properties; (6) reprocessing; and (7) safety.

- 168 (ANL-AFP--22) APPARATUS TO SIMULATE NUCLEAR HEATING IN ADVANCED FUELS. Wrona, B.J.; Galvin, T.M.; Johanson, E. (Argonne National Lab., IL (USA)). Oct 1976. Contract W-31-109-ENG-38. 20p. Dep. NTIS, PC A02/MF A01.

A direct-electrical-heating apparatus has been built to simulate in-reactor temperature gradients and heating conditions in both the mixed nitrides and carbides of uranium and plutonium. The apparatus has the capability for the investigation and direct observation of fuel-behavior phenomena that should significantly enlarge the data base on mixed carbides and nitrides at temperatures near and above their melting points. In addition to heating UC, results of proof tests showed that the apparatus has the capability to heat graphite, 30 vol % ZrC in graphite, B₄C control-rod pellets, and stainless steel.

- 169 (ANL-AFP--28) ADVANCED FUELS PROGRAM. QUARTERLY PROGRESS REPORT, JANUARY--MARCH 1976. Kittel, J.H. (comp.). (Argonne National Lab., IL (USA)). 1976. Contract W-31-109-ENG-38. 271p. AT.

Progress in studies on advanced LMFBR fuels is reported in the following areas: steady-

state irradiation tests, off-normal irradiation tests, performance modeling, design studies for FTR and CRBR, design studies for large plants, fuel properties, fuel chemical stability, environmental assessments, and safety.

- 170 (ANL-AFP--29) ADVANCED FUELS PROGRAM. QUARTERLY PROGRESS REPORT, APRIL--JUNE 1976. Kittel, J.H. (comp.). (Argonne National Lab., IL (USA)). 1976. Contract W-31-109-ENG-38. 245p. AT.

Progress in studies on advanced LMFBR fuels is reported in the following areas: steady-state irradiation tests, performance modeling, design studies for FTR and CRBR, design studies for large plants, fuel properties, fuel chemical stability, environmental assessments, and safety.

- 171 (ANL-AFP--30) BREEDING PERFORMANCE OF CARBIDE AND NITRIDE FUELS IN 2000 MWE LMFBRs: A PRELIMINARY REPORT. PART I. ASSUMPTIONS, CONSTRAINTS AND METHODOLOGY. Barthold, W.P. (Argonne National Lab., IL (USA)). Sep 1976. Contract W-31-109-ENG-38. 57p. Dep. NTIS, PC A04/MF A01.

The assumptions, constraints, and methodology used in the design and analysis of 2000 MWE LMFBRs using carbide and nitride fuel are presented. The assumptions used in the nuclear, thermal and mechanical designs are discussed together with geometry constraints, operational constraints, performance constraints, constraints coming from fabrication consideration and fuel bundle-duct interaction constraints. The basic calculational flow for system design studies is described, and the computer codes used in the analyses are briefly reviewed.

- 172 (ANL-AFP--31) BREEDING PERFORMANCE OF CARBIDE AND NITRIDE FUELS IN 2000 MWE LMFBRs. PART IV: IMPACT OF A PRESSURE DROP CONSTRAINT ON BREEDING PERFORMANCE AND PIN DIAMETER OPTIMIZATION IN 5000 MWT LMFBRs USING SODIUM BONDED CARBIDE FUELS. Lam, P.S. (Argonne National Lab., IL (USA)). Jul 1976. Contract W-31-109-ENG-38. 42p. Dep. NTIS, PC A03/MF A01.

Impact on pin diameter optimization and breeding performance from either a constant coolant velocity (25 ft/sec) constraint or a constant fuel bundle pressure drop (40 psi) constraint has been evaluated, and comparisons of these results are given. The 40 psi fuel bundle pressure drop was chosen not to imply a limit of pump technology, but because it corresponds to acceptable coolant velocities in the range of 25 ft/sec to 35 ft/sec for various pin diameters and linear heat ratings.

- 173 (ANL-AFP--32) SENSITIVITY OF POWER PEAKING ANALYSIS TO LARGE REACTOR CORE MODELING. King, M.J. (Argonne National Lab., IL (USA)). Oct 1976. Contract W-31-109-ENG-38. 47p. Dep. NTIS, PC A03/MF A01.

Various models of large LMFBRs, based on cylindrical and hexagonal geometries, are examined in regard to application to nuclear power peaking analysis. It is shown that the general behavior of power distributions during burnup in these large reactors implies that power shaping by control rod movement is desirable for minimizing the peaking factor. Due to current limitations of available three-dimensional computer codes, one- and two-dimensional models were used to test general assumptions of power shaping. However, it is demonstrated that these models are not appropriate for more accurate power peaking

analysis. Comparison of results for nuclear power peaking factors in cylindrical and hexagonal geometries shows that the use of annular rings of control poison in the cylindrical models does not provide a realistic representation of banks of control rod assemblies.

- 174 (ANL-AFP--33) ADVANCED FUELS PROGRAM. QUARTERLY PROGRESS REPORT, JULY--SEPTEMBER 1976. Kittel, J.H. (comp.). (Argonne National Lab., IL (USA)). 1976. Contract W-31-109-ENG-38. 199p. AT.

Progress in studies on advanced LMFBR fuels is reported in the following areas: steady-state irradiation tests, off-normal irradiation tests, performance modeling, design studies for FTR and CRBR, design studies for large plants, fuel properties, fuel chemical stability, environmental assessments, and safety.

- 175 (ANL-AFP--34) ADVANCED FUELS PROGRAM. QUARTERLY PROGRESS REPORT, OCTOBER--DECEMBER 1977. Kittel, J.H. (comp.). (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 196p. AT.

Progress is summarized in the following LMFBR mixed carbide and mixed nitride fuel development tasks: (1) design, surveillance, and evaluation of steady-state irradiation tests; (2) off-normal irradiation testing; (3) performance modeling; (4) design studies for FTF and CRBR; (5) design studies for large LMFBR plants; (6) fuel properties; (7) design of irradiation tests in FTF; (8) fuel chemical stability; and (9) safety studies. (DG)

- 176 (ANL-AFP--35) ADVANCED FUELS PROGRAM. QUARTERLY PROGRESS REPORT, JANUARY--MARCH 1977. Kittel, J.H. (comp.). (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 239p. AT.

Progress is summarized in the following LMFBR mixed carbide and mixed nitride fuel development tasks: (1) design, surveillance, and evaluation of steady-state irradiation tests; (2) off-normal irradiation testing; (3) performance modeling; (4) design studies for FTF and CRBR; (5) design studies for large LMFBR plants; (6) fuel properties; (7) design of irradiation tests in FTF; (8) fuel chemical stability; and (9) safety studies. (DG)

- 177 (ANL-AFP--36) RESPONSE MATRIX METHOD FOR LARGE LMFBR ANALYSIS. King, M.J. (Argonne National Lab., IL (USA)). Jun 1977. Contract W-31-109-ENG-38. 68p. Dep. NTIS, PC A04/MF A01.

The feasibility of using response matrix techniques for computational models of large LMFBRs is examined. Since finite-difference methods based on diffusion theory have generally found a place in fast-reactor codes, a brief review of their general matrix foundation is given first in order to contrast it to the general strategy of response matrix methods. Then, in order to present the general method of response matrix technique, two illustrative examples are given. Matrix algorithms arising in the application to large LMFBRs are discussed, and the potential of the response matrix method is explored for a variety of computational problems. Principal properties of the matrices involved are derived with a view to application of numerical methods of solution. The Jacobi iterative method as applied to the current-balance eigenvalue problem is discussed.

- 178 (ANL-AFP--37) CORE RESTRAINT PERFORMANCE OF LARGE HETEROGENEOUS REFERENCE CORES. de Paz, J.F.; Madell, J.T. (Argonne National Lab., IL (USA)). Jun 1978. Contract W-31-109-ENG-38. 77p. Dep. NTIS, PC A05/MF A01.

A preliminary study of the core restraint performance of heterogeneous cores was performed as part of the Large Heterogeneous Reference Core Study (LHRCS). The performance is based on satisfying four functional requirements: (1) to provide assembly alignment for control rod insertion and fuel handling; (2) to permit fuel handling within allowable insertion and withdrawal forces; (3) to maintain the core reactivity (from radial motion) within specified limits during normal and off-normal conditions; and (4) to prevent extensive duct-to-duct contact throughout the core. Six generic cases were selected to represent the range of core arrangements whose features greatly influence core restraint performance. Case 1 is a reference homogeneous core with temperature gradients only near the fuel-blanket interface. Case 2 is also a reference homogeneous core, but has fictitious opposing temperature gradients within the fuel region. Cases 3 and 4 are tightly-coupled heterogeneous cores with fuel and blanket in the center region, respectively. Cases 5 and 6 are loosely-coupled heterogeneous cores with fuel (Case 5) and blanket (Case 6) in the center region. The most important relative difference in the performance of homogeneous and heterogeneous cores is the extent of duct-to-duct contact in the core region. Three of the four heterogeneous cores have substantial duct-to-duct contact while neither of the homogeneous cores do. This performance characteristic of heterogeneous cores appear to be inherent in their design and not amenable to change by core modifications, etc. The features producing duct-to-duct contact should be kept in mind in the initial stages of laying out heterogeneous cores, and detailed core restraint analysis (i.e., 3-D) should be employed to investigate their core restraint performance in preliminary design studies.

Contract W-31-109-ENG-38. 45p. Dep. NTIS, PC A03/MF A01.

The consequences of carbide=fuel-coolant interactions (CFCI) resulting from overpower transients have been investigated with the PLUTO code (PLUTO1 version). The predictions are based on direct extrapolation from oxide-developed phenomenology. In addition the carbide results are compared to an "equivalent" oxide=fuel-coolant interaction (OFCI). The initial conditions for the fuel-coolant-interactions are based on predictions from whole core accident studies. The detailed calculations indicate that under these assumptions, using the oxide results as a standard of measure, the CFCI can be characterized by the following: (1) higher sodium temperatures and thus higher sodium vapor pressure in the interaction zone, (2) more rapid insertion of sodium voiding reactivity, (3) higher degree of z-symmetry (with respect to the rupture location) in the ejected fuel distribution, (4) more rapid growth of the interaction zone, and (5) an indication of the temporary shutting off of fuel ejection during the early stage of the interaction.

- 182 (ANL-AFP--46) ENGINEERING ANALYSIS OF MIXED CARBIDE FUELS FOR LARGE BREEDER REACTORS. de Paz, J.F.; Madell, J.T. (Argonne National Lab., IL (USA)). Jan 1978. Contract W-31-109-ENG-38. 51p. Dep. NTIS, PC A04/MF A01.

The results of several parametric studies of the performance of sodium-bonded carbide fuel assemblies are presented. The thermal hydraulic impact of changes in coolant velocity, core height, coolant temperature rise, spacer type and duct material is considered. An evaluation of the effect of these changes on the transient behavior, core restraint performance, flow-induced vibrations, and cladding lifetime is also given. The cladding thickness, inlet coolant temperature and spacer type are identified as the design features whose optimization appears most promising as a means of achieving an improved fuel performance.

- 179 (ANL-AFP--39) ADVANCED FUELS PROGRAM. QUARTERLY PROGRESS REPORT FOR THE PERIOD APRIL--JUNE 1977. Kittel, J.H. (comp.). (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 245p. AT.

Progress in the development of carbide/nitride LMFBR fuels is reported under the following headings: steady-state irradiation tests, off-normal irradiation tests, performance modeling, design studies for FTR and CRBR, design studies for large plants, fuel properties, design of irradiation tests in FTR, fuel chemical stability, and safety studies.

- 183 (ANL-AFP--47) ADVANCED FUELS DEVELOPMENT PROGRAM. QUARTERLY PROGRESS REPORT, OCTOBER--DECEMBER 1977. Kittel, J.H. (comp.). (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 223p. AT.

Progress is summarized in the following tasks related to development of LMFBR mixed carbide and mixed nitride fuels: (1) steady-state irradiation performance, (2) transient irradiation performance, (3) fabrication, (4) design and modelling, (5) fuels properties, (6) safety, and (7) environmental assessment. (DG)

- 180 (ANL-AFP--40) ADVANCED CARBIDE/NITRIDE FUELS PROGRAM. QUARTERLY PROGRESS REPORT FOR THE PERIOD JULY--SEPTEMBER 1977. Kittel, J.H. (comp.). (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 211p. AT.

Progress in the development of carbide/nitride LMFBR fuels is reported under the following headings: steady-state irradiation tests, off-normal irradiation tests, performance modeling, design studies for FTR and CRBR, design studies for large plants, fuel properties, design of irradiation tests in FTR, fuel chemical stability, and safety studies.

- 184 (ANL-AFP--51) ADVANCED FUELS DEVELOPMENT PROGRAM. QUARTERLY PROGRESS REPORT, JANUARY--MARCH 1978. Kittel, J.H. (comp.). (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 252p. AT.

Progress is summarized in the following tasks on mixed carbide and nitride fuel development for LMFBRs: steady-state irradiation performance, transient irradiation performance, fabrication, design and modeling, fuels properties, safety, and environmental assessments for mining and milling.

- 181 (ANL-AFP--41) COMPARISON OF CARBIDE AND OXIDE FUEL-COOLANT INTERACTIONS RESULTING FROM OVERPOWER TRANSIENTS. Mync, L.; Trease, H. (Argonne National Lab., IL (USA)). Dec 1977.

- 185 (ANL-AFP--55) ADVANCED FUELS DEVELOPMENT PROGRAM. QUARTERLY PROGRESS REPORT, APRIL--JUNE 1978. Kittel, J.H. (comp.). (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 240p. AT.

Progress is summarized in the following

- LMFBR mixed carbide and mixed nitride fuel development tasks: steady-state irradiation performance, transient irradiation performance, fabrication, design and modeling, fuels properties, safety, and environmental assessment.
- 186 (ANL-AFP--59) ADVANCED FUELS DEVELOPMENT PROGRAM. QUARTERLY PROGRESS REPORT, JULY-SEPTEMBER 1978. Kittel, J.H. (comp.). (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 216p. AT. Progress is summarized in the following mixed carbide and mixed nitride fuel development tasks: steady-state irradiation performance, transient irradiation performance, fabrication, design and modeling, fuels properties, and safety.
- 187 (ANL-AFP--63) ADVANCED FUELS DEVELOPMENT PROGRAM. QUARTERLY PROGRESS REPORT, OCTOBER-DECEMBER 1978. Kittel, J.H. (comp.). (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 248p. AT. Progress is summarized in the following LMFBR mixed carbide and mixed nitride fuel development tasks: (1) steady-state irradiation performance; (2) transient irradiation performance; (3) fabrication; (4) design and modeling; (5) fuel properties; (6) safety; and (7) environmental assessment.
- 188 (ANL-CT--77-6) HIGH-TEMPERATURE SHEAR-MODE TRANSDUCERS FOR LMFBR ULTRASONIC FLOWMETERS. Karplus, H.B. (Argonne National Lab., IL (USA)). Aug 1977. Contract W-31-109-ENG-38. 80p. AT. The design of an ultrasonic transducer for measuring sodium flow to 500°C in an LMFBR features a shear mode X-cut lithium niobate piezoelectric crystal mounted on a laminated coupling wedge. Acoustic transmission from the lithium niobate is effected by accurate surface flatness to 250 nm, an annealed platinum foil, and applied contact pressure of about 15 MPa (2000 psi). The pressure is applied through a stack of stainless steel foils which offer sufficient attenuation to the back-radiated energy to make its ultimate reflection and interference with the main pulse negligible. Sound energy is radiated into the liquid through mode conversion at the oblique interface at the inside surface of the pipe. The wedge shape of the coupling block eliminates standing waves in this block. This effect, combined with the close impedance match between lithium niobate and the steel wedge, yield very wide band widths (0.8-4 MHz). The response to a 100-ns pulse is one cycle of a distorted sine wave. Reverberation in the wedge due to residual reflection at the solid-liquid interface and grain boundary scattering is controlled by laminating the wedge, and forcing the stray interface reflection to traverse the laminations by machining the third wedge surface at a 60° angle. A piezoelectric element backed by a tuned tungsten carbide plate yielded a resonant transducer with a Q of 5. This unit was substituted for the low-temperature plastic wedge transducer supplied with a commercial clamp-on ultrasonic flowmeter. Preliminary measurements in a water loop have confirmed operation of the ANL developed high-temperature resonant transducer with commercial flowmeter electronics, requiring only a scale adjustment. Measurements of water flow using the highly damped transducer in conjunction with laboratory pulsers and timers, and special gates and trigger circuits yielded data needed for flowmeter design.
- 189 (ANL-CT--77-23) FINITE ELEMENT METHOD IN NONLINEAR DYNAMIC ANALYSIS OF AXISYMMETRIC SHELLS: THEORY AND COMPUTER PROGRAM. Hsieh, B.J. (Argonne National Lab., IL (USA)). Sep 1978. Contract W-31-109-ENG-38. 99p. TIC. Portions of document are illegible. The necessary equations for the equivalent nodal forces and the conditions necessary to simplify some of these equations are presented. A computer program is written in FORTRAN language that can treat uniformly distributed loadings. The modifications that are needed to treat other loadings are presented. This program is written in a nondimensional form so that any consistent units used for the input produces consistent units in the output.
- 190 (ANL-CT--77-25) MEASUREMENT OF LIQUID SODIUM FLOW IN THE SECONDARY SYSTEM OF THE EBR-II BY A NEUTRON ACTIVATION TECHNIQUE. Kehler, P.; Forster, G.A.; Price, C.C.; Sackett, J.I.; Curran, R.N. (Argonne National Lab., IL (USA)). Jun 1977. Contract W-31-109-ENG-38. 108p. AT. The liquid sodium flow velocity in the secondary system of the EBR-II was measured by a neutron activation technique. This test has verified the computer program used for the modeling of flow measurements by neutron activation techniques, has proven the feasibility of neutron activation methods for the in-situ calibration of flowmeters measuring liquid sodium flow through 30 cm pipes, and has provided calibration data for the EBR-II secondary system sodium flow instrumentation.
- 191 (ANL-CT--77-29) MEASUREMENT OF THE ACOUSTIC NOISE BACKGROUND IN THE EXPERIMENTAL BREEDER REACTOR II, UPPER PLENUM. Carey, W.M.; Bobis, J.P.; Gavin, A.P. (Argonne National Lab., IL (USA)). Dec 1977. Contract W-31-109-ENG-48. 199p. AT. Measurements of the acoustic background of an operating Liquid Metal Fast Breeder Reactor (LMFBR) have been performed over a seventeen month period in 1976 and 1977 at the Experimental Breeder Reactor II (EBR-II) in Idaho. These data are presented in terms of linear average spectra, autocorrelation and cross power spectral density for 0.2 to 5 kHz and 4 to 50 kHz analysis ranges. These data are shown to be weakly stationary. The data below 5 kHz were shown to be dominated by pump-induced structural vibrations and the excitation of acoustic resonances. The data above 5 kHz were shown to be the result of flow noise and the response of the reactor geometry to this flow noise. Acoustic resonances in the 4 to 50 kHz frequency region were closely spaced and produced a smeared response. The higher frequency data were also shown to be influenced by local flow effects and sensor mounting characteristics. A strong correlation of high frequency data with flow is presented, indicating that an appreciable portion of the high frequency data could be due to the action of each fuel subassembly acting as a subsonic jet. These measurements provide a basis for the acoustic surveillance of LMFBRs.
- 192 (ANL-CT--77-31) THERMAL BUOYANCY PHENOMENA IN A HORIZONTAL PIPE DURING A FLOW COAST-DOWN THERMAL-HYDRAULIC TRANSIENT (SCOPING STUDIES). Kasza, K.E.; Schmitt, R.C.; Sha, W.T. (Argonne National Lab., IL (USA)). Sep 1977. Contract W-31-109-ENG-38. 32p. AT. Initial problem scoping studies on thermal stratification effects in horizontal LMFBR piping during a pump coast-down cooling

- transient have been performed using the ANL COMMAX computer code. It is found that thermal buoyancy forces can create appreciable secondary flow and thermal effects. For the problem studied, donut shaped flow patterns develop in the pipe circular cross sections symmetrically on both sides of the pipe vertical symmetry plane; the patterns, for the case of a pipe inlet cooling transient, are produced by the sinking to the bottom of the pipe of the cooler fluid entering the pipe. In addition, a large longitudinal recirculation loop is established which convects warmer downstream fluid upstream in the pipe and greatly increases the pipe top-bottom temperature difference as well as causing the thermal-stratification-effects front to move faster through the pipe than would be indicated by the pipe inlet mean velocity. A pipe top-bottom stratification temperature difference in excess of 100°F (55.6°C) occurred which amounted to nearly 60% of the pipe-inlet-flow-transient temperature decline. This result contradicts the usual isothermal-front assumption employed in LMFBR system analysis codes. In general results are obtained which suggest that additional studies (both experimental and analytical) be undertaken to obtain information which can be directly applied to both design of LMFBR piping and assessment of thermal stratification effects under conditions of pump coast-down and natural circulation. Recommendations are also made as to the directions and focus of further research efforts.
- 193 (ANL-CT--77-33) ELECTROCHEMICAL OXYGEN METERS: STATE-OF-THE-ART, MARCH 1977. Epstein, L.F.; McKee, J.M. (Argonne National Lab., IL (USA)). Oct 1977. Contract W-31-109-ENG-38. 76p. AT.
- The present state-of-the-art of fabricating electrochemical oxygen meters, in the U.S. and abroad, for use in sodium-cooled reactor systems, particularly the LMFBR is summarized. It is concluded that there is a need for substantial improvement over the currently available meters, and that instruments that are more reliable and dependable can be developed and will be required in future advanced reactors.
- 194 (ANL-CT--77-34) NUBOW-2D INELASTIC: A FORTRAN PROGRAM FOR STATIC TWO-DIMENSIONAL STRUCTURAL ANALYSIS OF BOWED REACTOR CORES, INCLUDING EFFECTS OF IRRADIATION CREEP AND SWELLING. Cha, B.K.; McLennan, G.A.; Fulford, P.J. (Argonne National Lab., IL (USA)). Jun 1977. Contract W-31-109-ENG-38. 107p. AT.
- NUBOW-2D Inelastic is an extensive modification of the two-dimensional core-restraint analysis code NUBOW. Subroutines have been added to calculate time histories of inelastic changes in bowing shapes and forces induced by creep and swelling in reactor fuel assemblies (ducts). Based on the conventional beam theory assumption that plane sections remain plane, and using supplied radial and axial fluxes and temperatures, and creep and swelling correlations, the code calculates creep and swelling strains at axial and peripheral locations for each duct. These strains are numerically integrated to give corresponding bow shapes which are used as input pre-bow shapes for structural analysis by NUBOW. This procedure is performed on a time-incremental basis, giving a history of two-dimensional forces, displacements and strains for the ducts, and reactivity changes in the core.
- 195 (ANL-CT--77-35) VALIDATION OF THE COMPUTER PROGRAM PTA-1-RD. Hsieh, B.J. (Argonne National Lab., IL (USA)). Sep 1978. Contract W-31-109-ENG-38. 35p. AT.
- Experimental data on rupture discs generated by General Electric is used to validate the PTA-1-RD computer program. The experimental data was obtained by dropping a weight into a water filled pipe which was connected to a rupture disc at one end. Pressure transducers were located along the pipe to record the pressure produced as the resultant pulse moved through the system. One of the recorded pressure histories is used as the input disturbance to the PTA-1-RD program. The computed pressure histories from this program are then compared with the experimental data at other locations.
- 196 (ANL-CT--77-37) RESULTS FROM THE TESTING OF A TCE DC-14B AND THE ORNL FC-3 FISSION COUNTER-CHAMBERS IN THE EBR-II NITF AT TEMPERATURES UP TO 546°C. EBR-II RUN 82 (MAY 1976)--RUN 89 (AUGUST 1977). Taboas, A.L.; Yingling, G.E. (Argonne National Lab., IL (USA)). 14 Sep 1977. Contract W-31-109-ENG-38. 47p. AT.
- The performance of two high temperature fission counter-chambers in an LMFBR environment is evaluated and reported. The test results indicate that using current-sensitive preamplifiers, the ORNL FC-3 detector can be successfully operated as a long term LLFM at temperatures up to approximately 400°C, while the TCE DC-14B, S/N 7304-69 detector can be successfully operated as a long term, fixed position, wide range neutron flux monitor at temperatures up to approximately 565°C, in a gamma flux of 186 mC/Kg.s and a neutron flux greater than 1.1×10^{11} nv.
- 197 (ANL-CT--77-38) SUMMARY OF TEST RESULTS FROM THE TESTING OF A WESTINGHOUSE MODEL WX-30950 COMPENSATED IONIZATION CHAMBER IN THE EBR-II NITF J-1 THIMBLE AT TEMPERATURES UP TO 370°C (EXPERIMENT NI-3). Popper, G.F.; Elsbergas, V.J.; Yingling, G.E. (Argonne National Lab., IL (USA)). Sep 1977. Contract W-31-109-ENG-38. 39p. AT.
- A summary of the results obtained from the performance testing of a high temperature gamma compensated ionization chamber-cable assembly in a typical Liquid Metal Fast Breeder Reactor (LMFBR) nuclear environment at temperatures up to 370°C is presented. A brief description of the detector packaging used for installation into the reactor, the test equipment used and the data collection system is also included. Based upon the results contained in this report, the WX-30950 compensated ionization chamber tested could be used as a linear dc channel detector in an uncooled, fixed position LMFBR environment at temperatures up to 370°C.
- 198 (ANL-CT--78-1) CHF=INDUCED THERMAL OSCILLATIONS MEASURED IN AN LMFBR STEAM GENERATOR TUBE WALL. France, D.M.; Carlson, R.D.; Chiang, T.; Priemer, R. (Argonne National Lab., IL (USA)). Oct 1977. Contract W-31-109-ENG-38. 125p. AT.
- Thermal oscillations induced in a sodium heated steam generator tube wall as a consequence of critical heat flux (CHF) were experimentally measured. The tube was prototypical of the Clinch River Breeder Reactor (CRBR) plant design for the steam generator evaporator: length = 13.1 m, I.D. = 10.1 mm, O.D. = 15.9 mm. The tube was vertically orientated in the ANL Steam Generator Test Facility (SGTF) with water flowing vertically upwards inside and sodium flowing countercurrent in a surrounding

annulus. The SGTF test section and data analysis methods are presented in detail. The measured and calculated results of thermal fluctuation amplitude and thermal stress are presented showing the effects of sodium temperature and water pressure separately on the severity of the fluctuations.

- 199 (ANL-CT--78-2) CHARACTERIZATION OF THE TEMPERATURE FLUCTUATIONS GENERATED IN A THERMAL MIXING TEE (SODIUM VERSUS WATER BEHAVIOR). Kasza, K.F.; Colwell, W.S. (Argonne National Lab., IL (USA)). Nov 1977. Contract W-31-109-ENG-38. 88p. AT.

The temperature fluctuations occurring in 180° inlet-leg angle pipe tee thermal mixers with and without inlet reducer pipes are characterized. Sodium and water data from different sized tees have allowed delineation of the size and fluid medium modeling laws required for the extrapolation of reduced-scale water model temperature fluctuation data to prototype sodium mixers. The information presented is vital to the design of thermal-fatigue-free mixers and much of it has not been previously available to designers. The presented information on temperature fluctuations in pipe tees, in addition to furnishing much needed design information, also establishes that water testing of thermal mixers, with respect to modeling thermal-fatigue producing temperature fluctuations in prototype sodium mixers, has a built in conservativeness (i.e., $\Delta T_{\text{sub Na}} / \Delta T_{\text{sub H}_2\text{O}} < 1$). Finally, it is shown that thermal buoyancy effects can alter a thermal mixer's performance and that buoyancy must be correctly modeled in thermal mixing experiments.

- 200 (ANL-CT--78-3) NUMERICAL RESULTS OBTAINED FROM THE THREE DIMENSIONAL TRANSIENT SINGLE PHASE VERSION OF THE COMMIX COMPUTER CODE. Domanus, H.M.; Schmitt, R.C.; Sha, W.T. (Argonne National Lab., IL (USA)). Oct 1977. Contract W-31-109-ENG-38. 103p. (NUREG--0355). Dep. NTIS, PC A06/MF A01.

Results from the first three applications of the three-dimensional, transient, single-phase version of the thermal-hydraulics computer code, COMMIX, are presented. The applications include an LMFBR outlet plenum, a horizontal pipe, and a 19-pin LMFBR hexagonal fuel assembly. Each case is described and the results presented. All three applications use liquid sodium as the fluid. The plenum and pipe analyses are transient, while the fuel assembly involves a steady-state analysis.

- 201 (ANL-CT--78-5) MEASUREMENT OF THE SOUND PRODUCED DURING STEAM INJECTIONS INTO SODIUM. Claytor, T.N.; Carey, W.M.; Doolittle, R.D.; Fitzgerald, J.W. (Argonne National Lab., IL (USA)). Sep 1977. Contract W-31-109-ENG-38. 86p. AT.

The acoustic signature of a steam-into-sodium leak has been investigated. Characterization of the acoustic signal from a leak of steam into sodium is of interest because a steam leak in a liquid metal steam generator module has potential for dire economic consequences if allowed to grow into a larger-size leak by self-wastage and impingement damage. Passive acoustic monitoring of the steam generator module may have the capability to rapidly detect a steam-to-sodium leak in a liquid metal fast breeder reactor steam generator module. Data contained in the report clearly show that at higher frequencies (> 20 kHz) a larger signal-to-noise ratio was obtained than at frequencies below 10 kHz under conditions of half the maximum sodium flowrate

for a prototypic liquid metal steam generator. Characteristic of the steam-to-sodium leak was a form of acoustic energy release which occurs in intense bursts with frequency components of up to 100 kHz. This feature of the sodium-to-steam injection may enable detection of leaks with simplified instrumentation or improved response time by using steam leak signals in the frequency range above 20 kHz. At lower frequencies, the acoustic energy released was probably due to hydrogen bubble oscillation, thus substantiating previous suggestions that hydrogen bubble oscillation is the dominant source of sound at low frequencies (2 kHz).

- 202 (ANL-CT--78-6) GE/CRBRP/IHTS MIXING TEE MODEL PERFORATED PLATE VIBRATION TEST (IN SUPPORT OF GE/CRBRP:DRS 51.32). FINAL REPORT. Chung, H.; Hallie, H. (Argonne National Lab., IL (USA)). Sep 1977. Contract W-31-109-ENG-38. 71p. AT.

A series of vibration tests were performed on a quarter-scale model of CRBRP/IHTS mixing tee proposed by GE-FBRD to assess the vibration characteristics and the potential for flow-induced vibrations. The flow test conditions include simulated prototypic conditions using room-temperature water. The test results show that the fundamental natural frequency of the perforated plates is about 740 Hz in water and their displacements were less than 0.0025 mm (0.1 mils) RMS at the prototypic flowrate and at lower flowrates. The perforated plate response spectra were characterized by distinct, lightly damped resonances at all flowrates with high loop system static pressure. However, as the system pressure level was lowered the response spectra became less distinct, possibly due to the cavitation-type phenomena.

- 203 (ANL-CT--78-7) FINAL REPORT AND DESIGN GUIDE FOR CAVITATION-FREE LMFBR HYDRAULIC COMPONENTS. Cha, Y.S.; Huebotter, P.R. (Argonne National Lab., IL (USA)). Oct 1977. Contract W-31-109-ENG-38. 95p. AT.

A survey of experimental data on cavitation tests of various hydraulic components in water and sodium indicated that, for practical engineering applications, there is no significant difference between cavitation inception in water at room temperature and that in sodium at temperatures within the interest range of LMFBRs. It is concluded that water at room temperature can be used as a testing fluid to determine the cavitation inception threshold for sodium provided no deliberate action is taken to reduce the gas content in the test loop. Cavitation damage tests using venturis in water and sodium indicate that sodium is much more damaging than water under similar geometric and dynamic conditions. Under a typical loop-type LMFBR environment, cavitation damage is more severe in the cold leg than in the hot leg. Acoustic data from the cavitation tests of the FFTF primary sodium pump indicates that cavitation exists at NPSH values much greater than that corresponding to the 3% head-drop point.

- 204 (ANL-CT--78-8) INFLUENCE OF HETEROGENEOUS CORE GEOMETRY UPON OUTLET PLENUM MIXING. Howard, P.A. (Argonne National Lab., IL (USA)). Nov 1977. Contract W-31-109-ENG-38. 49p. AT.

LMFBR outlet plenum temperatures were simulated and recorded for three different heterogeneous core configurations. Core geometry included the CRBR Alternate Fuel Management Scheme (AFMS) case 9' and two modifications of the central assemblies.

Experiments were conducted at $1/10$ -scale where water was used as a test fluid. A total of 367 assemblies were simulated including fuel, inner blanket, outer blanket, and control assemblies. A scale model of the CRBR UIS was used in all tests and conditions close to Start-of-Cycle 1 were simulated. The steady-state temperatures and oscillations were recorded for 80 locations throughout the outlet plenum. Particular emphasis was placed upon chimney temperatures and leakage temperatures.

- 205 (ANL-CT--78-9) SODIUM FLOW CALIBRATION USING A PULSED NEUTRON ACTIVATION TECHNIQUE ON A 400MM (16 INCH) PIPE AT LMEC. Forster, G.A.; Kehler, P. (Argonne National Lab., IL (USA)). Nov 1977. Contract W-31-109-ENG-38. 33p. AT.

The velocity of liquid sodium in pipes can be measured by the pulsed neutron activation (PNA) technique without disturbing the pipe or removing insulation. A collimated beam of 14 MeV neutrons is directed against the sodium filled pipe. The resulting gamma activity is detected a suitable distance downstream. The velocity is determined from the spacing of source and detector, and the transit time of the activated sodium weighted for gamma decay. Data were taken on the 400 mm sodium system at LMEC at temperatures of 200, 400, and 560°C. The resulting velocities come within the 2 percent predicted uncertainty limit for the higher flow ranges when compared to the system's Venturi reference flowmeter. A low velocity measuring capability (.05 m/s) was also demonstrated.

- 206 (ANL-CT--78-10) CRITICAL HEAT FLUX TESTS AT LOW WATER FLOW AND 7.3 MPa: INTERIM REPORT. France, D.M.; Chiang, T.; Carlson, R.D. (Argonne National Lab., IL (USA)). Oct 1977. Contract W-31-109-ENG-38. 82p. AT.

Low flow critical heat flux (CHF) tests were performed in the ANL Steam Generator Test Facility (SGTF) employing sodium to boil water in a straight, vertically mounted, water tube under forced convection operation. The $2\frac{1}{4}$ Cr-1 Mo steel tube dimensions were: 13.1 m length, 10.1 mm I.D. and 15.4 mm O.D. Results of nine tests at 7.3 MPa water pressure and mass fluxes of 550, 270 and 140 kg/m² . s covering a broad range of critical heat flux, were selected for presentation in order to supply data on the earliest possible schedule. Parametric conditions at CHF are given including heat flux and quality. Measured and analyzed tube wall, thermal fluctuation data and thermal stresses are presented for each test. Only one test at the highest heat flux of 0.76 MW/m² and 270 kg/m² . s mass flux exhibited cyclic thermal stresses in the marginal range for meeting ASME requirements for 30 years lifetime. All other tests showed considerably lower stresses.

- 207 (ANL-CT--78-11) VIBRO-IMPACT RESPONSES OF A TUBE WITH TUBE--BAFFLE INTERACTION. Shin, Y.S.; Sass, D.E.; Jendrzeczyk, J.A. (Argonne National Lab., IL (USA)). Jan 1978. Contract W-31-109-ENG-38. 75p. Dep. NTIS, PC A04/MF A01.

The relatively small, inherent tube-to-baffle hole clearances associated with manufacturing tolerances in heat exchangers affect the vibrational characteristics and the response of the tube. Numerical studies were made to predict the vibro-impact response of a tube with tube-baffle interaction. The finite element method has been employed with a non-linear elastic contact spring-dashpot to model the effect of the relative approach between the tube and the baffle plate. The coupled

equations of motion are directly integrated with a proportional system damping represented by a linear combination of mass and stiffness. Lumped mass approach with explicit time integration scheme was found to be a suitable choice for tube-baffle impacting analysis. Fourier analyses indicate that the higher mode contributions to the tube response are significant for strong tube-baffle impacting. The contact damping forces are negligible compared with the contact spring forces. The numerical analysis results are in reasonably good agreement with those of the experiments.

- 208 (ANL-CT--78-12) TEMPERATURE-SENSITIVE ORIFICE: A PRE-TEST REPORT. Flynt, W.S.; Howard, P.A.; Carlson, R.D. (Argonne National Lab., IL (USA)). Nov 1977. Contract W-31-109-ENG-38. 31p. AT.

A device which is capable of maintaining a nearly constant blanket assembly exit temperature over several LMFBR fuel cycles is currently being investigated by the Components Technology Division of Argonne National Laboratory. This orifice will develop a pressure drop inversely proportional to temperature with the absence of any moving parts. Design and fabrication of a temperature-sensitive orifice scaled for the High Temperature Simulation Loop (HTSL) has also been completed. The proposed location of a prototypic orifice would be in the upper region of a LMFBR blanket assembly. This location is compatible with current LMFBR design.

- 209 (ANL-CT--78-13) COMPARISON AND VERIFICATION OF THE CORE RESTRAINT CODES NUBOW-2D, NUBOW-3D, AND PARABOW: LEVEL II. McLennan, G.A.; Moran, T.J.; Cha, B.K. (Argonne National Lab., IL (USA)). Nov 1977. Contract W-31-109-ENG-38. 108p. AT.

A series of ten problems was solved by each of the core restraint codes NUBOW-2D, PARABOW, and NUBOW-3D for purposes of verifying the calculational capabilities of the codes and comparing the solutions. The physical system modeled was the 121 assembly configuration of the Core Restraint Test Facility located at W-ARD. Problems with moving yokes and with varying temperatures were considered. The resulting calculated displacements of the individual assemblies and the contact forces between assemblies are reported. It is concluded that the codes accurately calculate the equilibrium forces and displacements as modeled, but that three dimensional effects cause significant difficulties in the equilibrium states for the different code models.

- 210 (ANL-CT--78-14) NEUTRON INDUCED CURRENT PULSES IN FISSION CHAMBERS. Taboas, A.L.; Buck, W.L. (Argonne National Lab., IL (USA)). Jan 1978. Contract W-31-109-ENG-38. 28p. Dep. NTIS, PC A03/MF A01.

The mechanism of neutron induced current pulse generation in fission chambers is discussed. By application of the calculated detector transfer function to proposed detector current pulse shapes, and by comparison with actually observed detector output voltage pulses, a credible, semi-empirical, trapezoidal pulse shape of chamber current is obtained.

- 211 (ANL-CT--78-15) MEASUREMENT AND CORRELATION OF CRITICAL HEAT FLUX IN A SODIUM HEATED STEAM GENERATOR TUBE. France, D.M.; Chiang, T.; Carlson, R.D.; Minkowycz, W.J. (Argonne National Lab., IL (USA)). Jan 1978. Contract W-31-109-ENG-38. 67p. AT.

Critical heat flux (CHF) experiments were performed in the Steam Generator Test Facility (SGTF) at Argonne National Laboratory (ANL) for application to Liquid Metal Fast Breeder Reactor (LMFBR) steam generators. A single, straight, vertical, full scale LMFBR steam generator tube was tested with water flow boiling upwards inside the tube heated by sodium flowing countercurrent in a surrounding annulus. The test section tube parameters were: inside diameter = 10 mm, outside diameter = 15.8 mm, material = $2\frac{1}{4}$ Cr - 1 Mo steel, and heated length = 13.1 m. Experiments were performed in the water pressure range of 7.0 to 15.3 MPa and the water mass flux range of 720 to 3200 kg/m² . s. The data exhibited two trends: heat flux independent and heat flux dependent. The CHF data from 398 tests in the SGTF are presented in summary along with an empirical correlation having a root mean square deviation of 14.7% from the data. The correlation is compared to other CHF data in the same parameter range.

- 212 (ANL-CT--78-16) PLENUM-2: A TWO-REGION OUTLET PLENUM MODEL. Howard, P.A.; Carbaño, J.J. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 93p. AT.

An experimental study to address the transient behavior of a generalized outlet plenum after a scram in a LMFBR has been conducted. Different conditions of flowrate, temperature, and geometries were used for three-exit loop-type reactor systems. Water was used as a fluid and a small-scale model plenum was used to simulate the various generalized configurations. Thermocouples were used to monitor the temperatures at 48 locations within the outlet plenum. The thermo-hydraulic behavior has been studied in detail and has been simplified as having four consecutive discrete stages. A computer model following these experimentally observed stages has been developed and calculations and results are compared.

- 213 (ANL-CT--78-19) NUBOW-3D (INELASTIC): A FORTRAN PROGRAM FOR THE STATIC THREE-DIMENSIONAL STRUCTURAL ANALYSIS OF BOWED REACTOR CORES INCLUDING THE EFFECTS OF IRRADIATION CREEP AND SWELLING. McLennan, G.A. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 176p. AT.

NUBOW-3D is a special purpose FORTRAN computer program for the three-dimensional static structural time history analysis of symmetric sectors of LMFBR core systems, subjected to thermal and/or mechanical bowing and time (fluence) dependent effects of irradiation-induced swelling and irradiation-enhanced creep. The program calculates the mechanical equilibrium forces and displacements of the core system treating the individual assemblies as conventional beam members. Time increments of creep and swelling strains are calculated for each assembly using specified material creep and swelling correlations and input values for assembly flux and temperature distributions. These strains are numerically integrated over each assembly to give inelastic beam curvatures and bowed shapes. Assembly dilation and contact forces in the core region are calculated for each time step. Detailed strain and fluence histories, contact forces and displacements are calculated and stored for each time step in the history. Using input reactivity worth values for individual assemblies, the structural reactivity changes for the reactor system are calculated for each time step. The NUBOW-3D structural model allows for axial variations in assembly geometry and different types of assemblies in the system.

Local flexibility and clearance gap may occur at all contact points. Contact stiffness is modeled as a single linear compression spring, acting after gap closure. Forces are assumed to act normal to contacting surfaces, so that friction and torsional effects are not included in the analysis.

- 214 (ANL-CT--78-20) RELEASE OF BUBBLES FROM SMALL LEAKS AT STEAM GENERATOR TUBESHEET WELDS. Skladzien, S.B.; McKee, J.M. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 19p. AT.

Helium was injected into a water-filled Lucite mockup of the CRBR steam generator tubesheet region to simulate the hydrogen bubbles formed by a small steam leak into sodium at a tube-to-tubesheet weld. With the tubesheet horizontal (lower tubesheet orientation), the bubbles escaped upward through the $\frac{1}{16}$ " (1.6 mm) annulus between the tubes and the 3" (76 mm) thick thermal baffle with no significant delay. With the tubesheet vertical (upper tubesheet orientation) the bubbles again escaped from the annulus and rose vertically through the gap between thermal baffle and tubesheet with minimal delay. The smallest leak that typically remains open, 10 mg H₂O/s, is expected to generate ample hydrogen to form a vigorous rising stream of bubbles. The total travel of these bubbles before dissolving is expected to be ample at the lower tubesheet and probably adequate at the upper (hotter) tubesheet to carry a local current of sodium rich in steam reaction products to the nearest leak detector. It is concluded that the thermal baffles and spacers will not significantly impede the prompt chemical detection of small steam leaks at tube-to-tubesheet welds in the CRBR hockey stick steam generator.

- 215 (ANL-CT--78-21) USER'S MANUAL FOR FINITE-ELEMENT COMPUTER CODE, CREEP-PLAST. Yang, C.I.; Zawadzki, S. (Argonne National Lab., IL (USA)). Jan 1978. Contract W-31-109-ENG-38. 55p. Dep. NTIS, PC A04/MF A01.

CREEP-PLAST is a finite element computer code developed for high temperature structural design of LMFBR vessels, components and core structures. It performs creep and plasticity calculation in two dimensional form under plane stress, plane strain or axisymmetric assumptions. The original code has been extensively modified to improve its efficiency.

- 216 (ANL-CT--78-22) FINITE-ELEMENT SOLUTION OF ADDED MASS AND DAMPING OF OSCILLATING RODS IN VISCOUS FLUIDS. Yang, C.I.; Moran, T.J. (Argonne National Lab., IL (USA)). Feb 1978. Contract W-31-109-ENG-38. 26p. Dep. NTIS, PC A02/MF A01.

A finite element analysis is presented for the cylindrical rods oscillating periodically in an incompressible viscous fluid. A system of discretized equation is obtained from the appropriate Navier-Stokes and continuity equations through Galerkin's process. The basic unknowns are velocity and pressure. A mixed interpolation method is used. The added mass and viscous damping coefficients which characterize the fluid reaction force due to the rods oscillation can be obtained through a line integration of stress and pressure around the circumference of the rods. For the special case of a cylindrical rod oscillating in a viscous fluid enclosed by a rigid, concentric cylindrical shell, the finite element solution agrees well with the analytical closed form solution, which, in turn, has been verified experimentally.

- 217 (ANL-CT--78-23) INTERIM ULTRASONIC SODIUM FLOW MEASUREMENTS ON THE EBR-II SECONDARY SYSTEM. Karplus, H.B.; Forster, G.A. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 45p. AT.

The ultrasonic flowmeter measures the flow in a pipe by determining the times of flight of an ultrasonic pulse travelling through the fluid in a diametral plane at an angle of about 55° to the pipe axis. The difference of the time of flight of a pulse travelling in the direction of the flow and a pulse travelling in the opposite direction yields a measure of the flow velocity. The average flight time of the two pulses is a function of the temperature from which a correction for pipe expansion and sound velocity may be derived. Flight times were measured with laboratory instruments and flow-rates calculated with a hand calculator to yield data for implementing a complete flowmeter using a microprocessor to give flow output directly. The differences between observed and calculated total ultrasonic flight times are well within experimental error. Absolute ultrasonic flow measurements are within the error band of other measuring systems.

- 218 (ANL-CT--78-24) PRESSURE TRANSIENT ANALYSIS IN PIPING SYSTEMS INCLUDING THE EFFECTS OF PLASTIC DEFORMATION AND CAVITATION. Youngdahl, C.K.; Kot, C.A.; Valentin, R.A. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 40p. Dep. NTIS, PC A03/MF A01.

Computational methods for analyzing pressure transients in the intermediate heat transport system of a sodium-cooled breeder reactor are being developed at Argonne National Laboratory. Because these systems typically operate at low pressures, thin-walled piping is used. Consequently, the pressure pulses produced by a sodium/water reaction in a steam generator or a pipe break may plastically deform sections of the piping, and rarefaction waves, because of the low operating pressure, may produce cavitation in the system. Both these phenomena have a large effect on the pressure pulses traversing the pipe network and, consequently, on the transient loading on major components. The computer program PTA-1, which includes the effect of plastic deformation of piping, and the computer program PTAC, which includes the effect of cavitation, have previously been validated using available experimental data. A new program, PTA-2, is being developed which combines the capabilities of PTA-1 and PTAC. Comparisons will be shown between PTA-2 predictions and the results of several experiments performed at Stanford Research Institute. In each of these experiments, a pressure pulse caused plastic deformation of a thin-walled pipe, producing a rarefaction wave which then produced a cavitating region in another pipe.

- 219 (ANL-CT--78-26) FTMOC-2D: A COMPUTER PROGRAM FOR TWO-DIMENSIONAL FLUID-TRANSIENT ANALYSIS BY THE METHOD OF CHARACTERISTICS. Shin, Y.W. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 133p. Dep. NTIS, MF A01.

Portions of document are illegible.

The computer program FTMOC-2D is a basic two-dimensional fluid-transient code that offers flexibility for application to a wide variety of problems. The flexibility is attained by requiring the user to provide a small part of the code to define the problem he wishes to solve. The basic numerical technique used is a

two-step explicit procedure based on the method of characteristics. Use of four linearized bicharacteristic relations yields an efficient computational procedure for both the Cartesian and axisymmetric two-dimensional problems. The basic scheme is suitable for a fluid-structure interaction problem. The program structure is described in detail. Ease in use of the program is demonstrated by a sample problem calculation. Graphical display capability is also developed by which the calculated result is easily visualized.

- 220 (ANL-CT--78-27) CRITICAL HEAT FLUX DATA WITH STRESS CALCULATIONS IN AN LMFBR STEAM GENERATOR TUBE. France, D.M.; Chiang, T.; Carlson, R.D.; Priemer, R.; Minkowycz, W.J. (Argonne National Lab., IL (USA)). Apr 1978. Contract W-31-109-ENG-38. 52p. AT.

Measured and calculated results of critical heat flux tests are presented from 406 experiments conducted in the ANL Steam Generator Test Facility. A full scale LMFBR steam generator tube was employed as the test section with an inside diameter = 10 mm, heated length = 13.1 m, and material = 2 1/2% Cr-1 Mo steel. Water was circulated vertically upwards through the tube and was heated by sodium flowing counter-current in a surrounding annulus. Complete parametric results are presented for all tests, and additional measured and calculated results are included for thermal fluctuations and thermal stresses in the tube wall adjacent to the transition boiling region for 139 of the tests.

- 221 (ANL-CT--78-30) OUTLET PLENUM MIXING FOR TRANSIENT OVERPOWER CONDITIONS OF A ONE-EXIT NOZZLE LMFBR. Howard, P.A. (Argonne National Lab., IL (USA)). Apr 1978. Contract W-31-109-ENG-38. 70p. Dep. NTIS, PC A04/MF A01.

Two types of transient tests were employed to model a one-exit nozzle LMFBR outlet plenum. Water was used as a test fluid in the simulation of constant flowrates, Transient Overpower (TOP) conditions. In the first test, simulated fuel flow was 85% and blanket flow was 15%, whereas in the second test, the fuel flow was 100%. This allowed the assessment of the mitigating effects of blanket flow upon the exit nozzle temperature transient. The flow field was clearly three-dimensional, and a less active, though not stagnant, region was observed diametrically opposite the exit nozzle. During steady state, oscillations above the fuel-blanket interface were found to be small. This is attributed to the existence above the reactor core of a recirculating flow field, which served as an effective mixing agent. A simple lumped-parameter model, EXITI, was developed to simulate TOP transient conditions for the test with both fuel and blanket flows. The predicted temperature profiles for various regions in the plenum were in good agreement with the experimental profiles, except for the region immediately above the reactor blanket. In devising the computer model, the temperature in this region was assumed to remain constant throughout the transient. However, this constant temperature did not prevail owing to the mixing that occurred in this region as a result of the recirculating flow field above the reactor core. The computer model can be readily modified to take into account the mixing due to this recirculation. In the test without blanket flow, good agreement between predictions and data was again obtained. In comparing results of the two tests, it was found that the blanket flow had only a small mitigating influence on the transient at the exit nozzle. The computer

model can easily be extrapolated to reactor conditions.

- 222 (ANL-CT--78-32) PTA-1-RD: A COMPUTER PROGRAM FOR ANALYSIS OF PRESSURE TRANSIENTS IN HYDRAULIC NETWORKS INCLUDING THE EFFECTS OF NONINSTANTANEOUS RUPTURE DISCS. Hsieh, B.J. (Argonne National Lab., IL (USA)). Sep 1978. Contract W-31-109-ENG-38. 73p. AT.

The one dimensional method of characteristics pressure transient analysis computer program, PTA-1 can treat complex piping networks and includes a variety of junction types and boundary nodes. Among these boundary nodes is a rupture disc representation. However, the disc is assumed to fail instantaneously when subjected to a pressure preceding some prespecified value, usually the static buckling pressure of the disc. A nonlinear finite element method is applied to compute the transient response of the disc. The disc's response is then fed back to the PTA-1 program by using the Newton-Raphson iteration technique. The resulting new computer program, PTA-1-RD, retains all the capabilities of the PTA-1 code and, in addition, treats the noninstantaneous rupture disc.

- 223 (ANL-CT--78-33) THERMODYNAMIC THEORY OF CAVITATION NUCLEI IN DILUTE LIQUID-GAS SOLUTIONS. Cha, Y.S. (Argonne National Lab., IL (USA)). Jun 1978. Contract W-31-109-ENG-38. 39p. Dep. NTIS, PC A03/MF A01.

The stability of a spherical bubble in a two-component two-phase system is examined by employing the thermodynamic theory of dilute solutions. It is shown that a bubble can remain in a state of stable equilibrium provided that the ratio of the total number of moles of the solute to the total number of moles of the solvent in the system is not extremely small and that the system pressure falls between an upper bound (dissolution limit) and a lower bound (cavitation limit). The results of the analysis provide a theoretical basis for the persistence of microbubbles in a saturated liquid gas solution. Thus to a certain extent, the results also help to resolve the dilemma that exists in the field of cavitation due to (1) the necessity of postulating the existence of microbubbles; and (2) the lack of theoretical justification for the persistence of such bubbles in a liquid.

- 224 (ANL-CT--78-34) COUPLED SHELL-SIDE AND TUBE-SIDE HEAT TRANSFER MODEL FOR HEAT EXCHANGERS. Oras, J.J.; Sha, W.T.; Kao, T.T.; Chou, S.L.; Cho, S.M. (Argonne National Lab., IL (USA)). May 1978. Contract W-31-109-ENG-38. 23p. AT.

A heat transfer model coupling the heat transfer of the shell- and tube-side fluids of a sodium-to-sodium heat exchanger is developed. In addition, a difference scheme is suggested for use in computer simulations. Nonuniform tube-side flow distribution is incorporated in the model. Future extension of the model to steam generator applications is made possible by the use of energy equations written in terms of enthalpy.

- 225 (ANL-CT--78-39) CRITICAL HEAT FLUX MEASUREMENTS WITH SODIUM HEATING AND LOW WATER FLOWRATE. France, D.M.; Chiang, T.; Carlson, R.D.; Minkowicz, W.J.; Priemer, R. (Argonne National Lab., IL (USA)). Jul 1978. Contract W-31-109-ENG-38. 37p. AT.

Seventy critical heat flux (CHF) experiments were performed in the Steam Generator Test

Facility (SGTF) employing a straight vertical steam tube of 13.1 m length, 10 mm inside diameter, heated by flowing sodium. The water mass flux was relatively low ranging from 0.1 to 0.6 Mg/m².s, and the majority of the data were obtained at a water pressure of 7.3 MPa. The data are presented in measured and reduced form for each test. The critical quality and thermal stress in the tube wall at CHF are compared to extrapolations of previously derived empirical correlation equations.

- 226 (ANL-CT--78-40) UK/USA CORE RESTRAINT EXCHANGE STUDY NUBOW-3D RESULTS. McLennan, G.A. (Argonne National Lab., IL (USA)). Jul 1978. Contract W-31-109-ENG-38. 73p. AT.

Core restraint structural analyses were performed for a model of the National Core Restraint Test Facility (NCRTF) for a series of loading conditions. These calculations were performed using the computer code NUBOW-3D, to form a basis for comparison with results to be obtained through a cooperative agreement with the United Kingdom, from the UK code CRAMP. This UK core restraint code includes a friction capability and the results of this comparison will be used to assess the effects of friction on US core restraint analysis and design. The report summarizes the NUBOW-3D modeling of the NCRTF to be compatible with the UK code and presents the (frictionless) NUBOW-3D results for a series of selected loadings, including compaction by movable yokes at the above-core-load-plane, and an assumed thermal rise to power for a specified temperature distribution.

- 227 (ANL-CT--78-42) FINITE-ELEMENT CODE FOR COMPUTING ADDED MASS COEFFICIENTS. Yang, C.I. (Argonne National Lab., IL (USA)). Sep 1978. Contract W-31-109-ENG-38. 21p. AT.

In this report, the effect of the surrounding fluid which is termed as added mass upon the accelerating structural components is outlined. If the surrounding fluid is assumed to be inviscid, the governing equation can be expressed in terms of pressure and reduced to elliptic type. A numerical approximation of the solution based on a previous report has been formulated. For a special case, such as a set of concentric cylinder with fluid in the gap, the results agree well with the analytical solution. This computer code was also used to compute the matrix of added mass coefficients of an array of hexagonal cans surrounded by inviscid fluid.

- 228 (ANL-CT--78-43) HYDRODYNAMIC FORCES ON A GROUP OF TUBES SUBJECTED TO CROSSFLOW. Chen, S.S.; Lin, W.H. (Argonne National Lab., IL (USA)). Jul 1978. Contract W-31-109-ENG-38. 63p. AT.

A complete solution to the problem of crossflow-induced vibrations of tube banks is still unattainable. The major task is the characterization of flow noises and motion-dependent forces. This includes the understanding of formation of organized vortex systems, unsteady fluid forces acting on the tubes, turbulence characteristics within tube arrays, and interaction of tube motion and fluid field. The objective of this report is to present a two-dimensional, potential-flow solution for the fluid inertia force, hydrodynamic damping force, and fluidelastic force of tube arrays arranged in an arbitrary pattern. Based on the result, a mathematical model for crossflow induced of tube banks is presented.

- 229 (ANL-CT--78-44) EXPERIMENTS ON

FLUIDELASTIC INSTABILITY IN TUBE BANKS SUBJECT TO LIQUID CROSS FLOW. PART I. RECTANGULAR ARRAYS. Chen, S.S.; Jendrzejczyk, J.A.; Lin, W.H. (Argonne National Lab., IL (USA)). Jul 1978. Contract W-31-109-ENG-38. 256p. AT.

The objective of the work reported is to provide an experimentally validated mathematical model for design evaluation of curved tube banks to avoid detrimental flow-induced vibrations. An extensive test program is being carried out to establish the critical flow velocity as a function of various system parameters. The test program is divided into two phases: (1) rectangular tube arrays and (2) triangular and mixed tube arrays. Testing of the rectangular tube arrays has been completed; the report presents the results for the first-phase tests. Critical flow velocities for six tube arrays with different spacing, mass ratio, damping, and detuning are established. Based on the experimental data, a stability map is prepared; this is useful in design to avoid detrimental fluidelastic instability. Based on the results, several general conclusions are drawn: (1) the generally accepted correlation does not correlate the data well, (2) detuning has beneficial effects on tube stability, (3) in the parameter ranges tested, tube spacings in the transverse and longitudinal directions do not have a significant effect on tube stability, (4) the instability modes are predominantly associated with the lift-direction motions, (5) there are multiple values of critical flow velocities associated with different modes of instability, (6) fluidelastic mechanism interacts with vortex shedding in some cases, and (7) damping stabilizes tube arrays and affects the on-set of instability.

230 (ANL-CT--78-47) INITIAL CONSIDERATIONS ON THE INFLUENCE OF THERMAL BUOYANCY ON HEAT EXCHANGER PERFORMANCE (RELEVANT GEOMETRIC, OPERATIONAL, THERMAL-HYDRAULIC PHENOMENA AND MODELLING CONSIDERATIONS). Kasza, K.E. (Argonne National Lab., IL (USA)). Aug 1978. Contract W-31-109-ENG-38. 130p. AT.

An initial study of the propensity for thermal buoyancy to alter the performance of the current generation of LMFBR intermediate heat exchangers under low flow operation has been performed. The thermal buoyancy related phenomena which can possibly occur in the various geometric subregions of the IHX and the appropriate modelling parameters for many of these phenomena have been delineated. The extent of the knowledge available pertaining to the various phenomena has been determined through a quite extensive literature review. In general there is a weakness in the general understanding of thermal buoyancy effects in complex flow and heat transfer equipment.

231 (ANL-CT--79-1) MIXING STUDIES FOR GEOMETRIC VARIATIONS OF THE CRBR OUTLET PLENUM. Howard, P.A.; Carbajo, J.J. (Argonne National Lab., IL (USA)). Oct 1978. Contract W-31-109-ENG-38. 59p. AT.

A set of eight transient mixing tests was performed with five geometric variations of the CRBR outlet plenum. These results are compared to data from an earlier study where the prototypic CRBR outlet plenum was modeled. Water was used as a test fluid and similitude was based upon the Richardson number at 1/10 scale. A model of the CRBR reference core with both fuel and blanket assemblies was utilized throughout. The results of two different scram transients for similar geometry are compared, and the question of simulation accuracy is addressed. In addition, experimental results are compared to predictions from the PLENUM-2

computer code which provided good agreement with the representative results.

232 (ANL-CT--79-3) FLUID FORCES ON RODS VIBRATING IN FINITE LENGTH ANNULAR REGIONS. Mulcahy, T.M. (Argonne National Lab., IL (USA)). Jun 1978. Contract W-31-109-ENG-38. 37p. AT.

Approximate expressions for the fluid forces acting on a central, rigid rod translating periodically in a finite length annular region of confined fluid are derived from the Navier Stokes equations for a range of geometric and fluid parameters where viscous damping is significant. Based on the derived forces, lumped added mass and linear dashpot modeling of an annular gap support typically found in nuclear reactors are employed to predict the fundamental frequency and modal damping of a single beam. Test methods and results for the same beam are presented which indicate the force expressions are applicable for small fluid motions.

233 (ANL-CT--79-4) THERMAL-HYDRAULICS DESIGN GUIDE FOR AN ADVANCED DESIGN THERMAL MIXER (INCLUDING DEVELOPMENT/TESTING DETAILS). Kasza, K.E.; Bobis, J.P. (Argonne National Lab., IL (USA)). Oct 1978. Contract W-31-109-ENG-38. 88p. AT.

An Advanced Design Thermal Mixer concept has been developed and performance tested. The mixer combines two fluid streams by utilizing a coaxial perforated pipe insert which allows flow from an outer annular region to jet radially into an inner pipe through a large number of holes and combine with the second stream. Detailed mixer performance information (both experimental and analytical) has been generated on temperature fluctuation peak-to-peak amplitude, associated temperature fluctuation frequency, and pressure changes. The dependence of the preceding measures of mixer performance on thermal-hydraulic conditions and mixer geometry has been delineated. The peak-to-peak temperature fluctuation amplitudes ($\Delta T/\text{sub FLUCT/}$) occurring at the mixer primary pressure boundary pipe possess a maximum which is a function of geometry and thermal-hydraulic conditions. A general correlation in terms of three basic mixing mechanisms has been developed which relates $\Delta T/\text{sub FLUCT/}$ to all pertinent parameters. The mixer concept is shown to be capable of achieving rapid thermal mixing for which $\Delta T/\text{sub FLUCT/}$ is less than 20% of the inlet-leg two stream temperature difference. In contrast conventional pipe tee mixers previously tested readily experienced $\Delta T/\text{sub FLUCT/}$ of order 90% of the inlet-leg temperature difference. The low level of $\Delta T/\text{sub FLUCT/}$ achievable in the subject mixer design qualifies it as thermal-fatigue free mixer from LMFBR service.

234 (ANL-CT--79-5) PRELIMINARY TEST PLAN FOR SEISMIC FEATURE TESTS. Moran, T.J. (Argonne National Lab., IL (USA)). Oct 1978. Contract W-31-109-ENG-38. 11p. AT.

The purpose of this document is to describe a series of seismic feature tests to be conducted by the Components Technology Division of Argonne National Laboratory for the Reactor Research and Technology Division of the U.S. Department of Energy. This experiment is part of the National Core Mechanics and Configuration Performance Plan Seismic development program under Task C, Feature Tests. In particular, Phase I address tasks C.1.1, gap impact phenomena tests, C.1.2, impact parameter variations, and C.2, inertial

coupling, damping and effective mass effects. Later phases of the test series may address tasks C.1.3, impact damage tests and C.3, dynamic friction effects. This series of tests is intended to qualify mathematical models of fluid/structure interaction and structures impacting in fluid for systems analogous to those occurring in fast reactor core seismic response. The seismic analysis of fast reactor core structures is a complex problem involving fluid structural coupling and impact in a three-dimensional structure; it requires extensive simplifying assumptions to obtain a solvable mathematical model. Seismic testing of reactor core models is also a complex and expensive effort. The seismic feature tests provide the intermediate step of experimental verification of the analysis tools and assumptions in simplified models representative of various features of core seismic response.

- 235 (ANL-CT--79-06) DYNAMIC RESPONSE OF CIRCULAR TUBES SUBJECTED TO LIQUID CROSS FLOW. Chen, S.S.; Jendrzeczyk, J.A.; Wambsganss, M.W. (Argonne National Lab., IL (USA)). Oct 1978. Contract W-31-109-ENG-38. 77p. AT.

The work reported herein was performed as part of the base technology activity under Flow Induced Vibration Program (189a No. CA054) sponsored by DOE/RRT. The overall objective of the activity is to develop new and/or improved, experimentally-validated analytical methods and guidelines for designing breeder reactor components to avoid detrimental flow induced vibration. This report describes three series of experimental investigations of circular tubes subjected to liquid cross flow: (1) a single tube; (2) a pair of tubes with one in the wake of the other; and (3) two tubes in a plane normal to the flow stream. The tube response characteristics, including natural frequencies, dampings, displacements, and vibration orbits are studied in detail. The results of this study provide some additional insight into the response characteristics of circular tubes in liquid flow, such as flow-velocity dependent damping and "lock-in" region of tube motion in the drag direction.

- 236 (ANL-CT--79-11) SOFTWARE DEVELOPMENT FOR THE MIXING COMPONENTS TEST FACILITY. Bobis, J.P.; Colwell, W.S. (Argonne National Lab., IL (USA)). Nov 1978. Contract W-31-109-ENG-38. 71p. AT.

The first set of software was developed and performance tested for the 16 bit minicomputer system at the Mixing Components Test Facility. The software developed for the analog/digital converter allows 32 channels of analog signals to be scanned in less than 4 ms with a minimum scan interval of 20 ms. For the analysis of the data in the time domain additional software was developed that calculated the mean, rms, maximum and minimum signal levels, and listed instantaneous values of chosen signals. To assist in determining the frequency associated with a temperature fluctuation, software was generated and tested successfully to measure the signal power spectral density and autocorrelation. For two temperature signals that can be related by a time delay the autocorrelation software was extended to operate in a cross correlation mode. The time and frequency domain analysis was supplemented with analysis in the amplitude domain when software was completed for determining signal probability density histograms. All software was evaluated and proven to be correct with the use of benchmark analytic functions and by measuring the analog signal with hardware that included rms-mean voltmeters, storage oscilloscopes, and spectrum analyzers.

- 237 (ANL-CT--79-14) POST-CHF HEAT TRANSFER AND TRANSITION BOILING CHARACTERISTICS IN AN LMFBR STEAM GENERATOR TUBE. Chiang, T.; France, D.M.; Carlson, R.D. (Argonne National Lab., IL (USA)). Dec 1978. Contract W-31-109-ENG-38. 139p. AT.

One hundred thirty-five post-critical heat flux (post-CHF) experiments were performed in the Steam Generator Test Facility (SGTF) employing a straight, vertical steam tube of 13.1 m length and 19 mm inside diameter that was heated by flowing sodium. The water pressure ranged from 7.0 to 15.3 MPa, and the water mass flux ranged from 0.72 to 3.2 Mg/m².s. The data and a correlation equation are presented. Also presented are the results of analyses of transition boiling region characterization which show that the previously observed heat flux dependent and heat flux independent trends of critical heat flux data are a consequence of a change of the hydrodynamic structure of the two-phase flow.

- 238 (ANL-CT--79-16) REVIEW ON FLUID FORCES ON CIRCULAR CYLINDERS IN CROSS FLOW. Nahavandi, A.N.; Chen, S.S. (Argonne National Lab., IL (USA)). Jan 1979. Contract W-31-109-ENG-38. 68p. AT.

The objective of the study is to recommend the most promising numerical digital-computer-oriented methods for the computation of fluid forces on a single rigid cylinder or an array of rigid cylinders. The scope of the work specifically includes: (1) performing an extensive literature search on the analytical, numerical and other solution techniques applicable to this problem; (2) comparing the advantages and disadvantages of the available solution techniques; and (3) recommending the most promising numerical techniques and computer codes for the computation of hydrodynamic loading on cylinders in cross flow.

- 239 (ANL-CT--79-17) PLENUM=2A: A PROGRAM FOR TRANSIENT ANALYSIS OF LMFBR OUTLET PLENUMS. Howard, P.A.; Carbajo, J.J. (Argonne National Lab., IL (USA)). Feb 1979. Contract W-31-109-ENG-38. 52p. AT.

An improved version of the PLENUM=2 code, PLENUM=2A, has been developed to predict the thermal conditions in an LMFBR outlet plenum after a reactor scram. Both the old and the improved versions divide the transient into three consecutive stages and consider only two mixing regions within the outlet plenum. Modifications of the code as reported here include elimination of the three second delay between Stage 1 and Stage 2, a simplification of the transition stage which occurs between complete mixing and the stratification phase, and an improved hot-cold interface rise-rate correlation. In most cases, PLENUM=2A has better agreement with available data than does PLENUM=2.

- 240 (ANL-CT--79-18) TESTING OF STEAM GENERATOR LEAK DETECTORS ON THE HIGH PRESSURE TEST LOOP: INTERIM REPORT. Koehl, E.R.; Smith, F.A.; McKee, J.M. (Argonne National Lab., IL (USA)). Jul 1979. Contract W-31-109-ENG-38. 63p. AT.

A small pumped loop was built and is being used to test LMFBR steam generator leak detectors and calibration equipment in sodium at 925°F and 180 psi, with excursions to 325 psi (planned CRBR operating conditions). Six hydrogen meters, four oxygen meters and a hydrogen injection coil have completed a year of operation in sodium with no mechanical

failures. The calibration stability of the hydrogen meters has not been as good as expected and is the subject of continuing investigation. The hydrogen injection coil was found to be correctly sized for its intended function, on-line calibration of hydrogen meters in leak detector modules. Sodium hydroxide injections are being used to calibrate both hydrogen and oxygen meters. Essentially all of the hydrogen and oxygen in the hydroxide is promptly detectable at 925°F. Two nickel membrane designs and two ion-pump cathode materials are being tested on the six hydrogen meters.

- 241 (ANL-CT--79-32) ACOUSTIC RADIATION AND SCATTERING FROM A GROUP OF CIRCULAR CYLINDERS. Lin, W.H.; Chen, S.S. (Argonne National Lab., IL (USA)). Apr 1979. Contract W-31-109-ENG-38. 76p. AT.

Acoustic waves caused by the motion of a group of circular cylinders in an infinite compressible fluid medium have been investigated. The linear two-dimensional acoustic wave equation is solved in terms of a series of cylindrical wave functions associated with the circular coordinates of each cylinder. To satisfy the boundary condition at a particular cylinder, all cylindrical wave functions are transformed to the local coordinates of that cylinder. The resulting equations are a system of linear algebraic equations for the undetermined constants, which are solved numerically by a digital computer. The velocity potential, pressure, and force acting on each cylinder are then obtained in terms of these constants.

- 242 (ANL-CT--79-35) SWAAM PREDICTION OF LLTR SERIES-II A-1A TEST. Shin, Y.W.; Lin, H.C.; Eichler, T.V.; Hsieh, B.J. (Argonne National Lab., IL (USA)). May 1979. Contract W-31-109-ENG-38. 59p. AT.

This report describes the pre-test prediction of the nonreactive nitrogen injection Test A-1a of the LLTR Series-II Test sequence. The test conditions resemble the reactive Test A-2 subcooled water injection at 1700 psia except that the tube-side pressure is increased to 2000 psia to insure activation of the relief system. The relief system activates when the rupture disk fails.

- 243 (ANL-CT--79-36) MIXING IN A POOL-TYPE PLBR AFTER A SCRAM. Howard, P.A.; Haried, J.A. (Argonne National Lab., IL (USA)). Jun 1979. Contract W-31-109-ENG-38. 116p. AT.

Transient flow patterns were simulated for eleven geometric variations of a pool-type PLBR. The effects of coolant density changes due to a reactor scram were simulated using water in a 1/15-scale model of the hot and cold pools. The influence of Intermediate Heat Exchanger (IHx) inlet height, gap beneath the Upper Internal Structure (UIS), and UIS exit height on coolant flow were studied in the hot pool. Transients with and without pump trips were included in this study. Dye injection allowed visual observation of the flow patterns within the hot pool. The results for the hot pool were compared to an existing computer model, Plenum-2A. In addition, the cold pool was analyzed during simulated upshock transients with various primary pump inlet heights.

- 244 (ANL-CT--79-40) SCALE MODELING OF FINITE LENGTH FLUID ANNULAR GAPS FOR FLOW-INDUCED VIBRATION TESTING. Mulcahy, T.M.; Miskevics, A.J. (Argonne National Lab., IL

(USA)). Jun 1979. Contract W-31-109-ENG-38. 30p. AT.

Possible scale model distortions for structural vibration testing are discussed. Conditions for which fluid energy dissipation is expected to be larger in a reduced scale model than in the prototype are defined. This unconservative and sometimes undesirable distortion is confirmed by testing of a vibrating rod surrounded by a finite length fluid filled annulus. A theory for estimating the magnitude of the distortion is presented.

- 245 (ANL-CT--79-45) EXPERIMENTS ON FLUIDELASTIC INSTABILITY IN TUBE BANKS SUBJECT TO LIQUID CROSS FLOW. PART II. MIXED ARRAYS. Chen, S.S.; Jendrzejczyk, J.A.; Lin, W.H. (Argonne National Lab., IL (USA)). Jul 1979. Contract W-31-109-ENG-38. 298p. AT.

The work reported was performed as part of the design method development activity for curved tube banks under the Flow Induced Vibration Program included in US DOE/RRT Work Package: Engineered System and Components; Task: General Engineering Support (Task No. 00305). The objective is to provide an experimentally validated mathematical model for design evaluation of curved tube banks to avoid detrimental flow-induced vibrations. An extensive test program was carried out to establish the critical flow velocity as a function of various system parameters. The test program was divided into two phases: (1) rectangular tube arrays; and (2) mixed tube arrays. The results from the second-phase tests are presented.

- 246 (ANL-CT--79-46) FIC: A FINITE ELEMENT CODE FOR CALCULATING ADDED MASS AND DAMPING COEFFICIENTS. USER'S MANUAL. Batill, S.M. (Argonne National Lab., IL (USA)). Jul 1979. Contract W-31-109-ENG-38. 60p. AT.

The purpose of this report is to provide a user's manual for a finite element code, written by C. I. Yang of ANL. The code is capable of calculating the hydrodynamic reactions on multiple interacting bodies undergoing simple harmonic motion. This report will document input requirements and discuss the program output. A sample data set and program listing are also included. It should be noted that the program has not been prepared for production applications but is intended for use as a research tool.

- 247 (ANL-CT--79-47) PRELIMINARY STUDY OF FLUID EFFECTS ON REACTOR CORE SEISMIC RESPONSE. Batill, S.M. (Argonne National Lab., IL (USA)). Jul 1979. Contract W-31-109-ENG-38. 34p. AT.

The dynamic response during a seismic event of a liquid metal fast breeder reactor core may be influenced significantly by the fluid contained between the fuel assemblies. This report documents a preliminary study designed to highlight the dominant fluid effects on the dynamic response of a related idealized model. The model consists of a simple two-dimensional rectangular channel which contains five rigid blocks with square cross-sections. Fluid added mass and damping coefficients were calculated using a finite element model of the fluid region. These effects were incorporated into the SCRAP computer code and position-time histories due to simple harmonic base motion were calculated for the idealized core.

- 248 (ANL-CT--79-48) SCRAP: INPUT, OUTPUT, AND SOURCE LISTINGS. Moran, T.J. (Argonne National Lab., IL (USA)). Jul 1979.

Contract W-31-109-ENG-38. 108p. AT.

This report describes the FORTRAN IV computer code, SCRAP, written to aid the analysis of the seismic response of fast reactor core structures. Flow diagrams for the code are given, the code input is described and examples listed, and the source listing is given. Comparison reports describe the theoretical basis of the SCRAP code and provide guidance in modeling fast reactor core structures with the code. SCRAP is a research tool and changes to the code are expected so this report should be considered interim.

- 249 (ANL-CT--79-49) SCRAP: THEORY AND MODELING CONSIDERATIONS. Moran, T.J. (Argonne National Lab., IL (USA)). Jul 1979. Contract W-31-109-ENG-38. 28p. AT.

The theoretical basis for the FORTRAN IV computer code SCRAP is presented. Guidance for modeling fast reactor core structures with the code is included. Companion reports describe the FORTRAN source and input format and the theoretical basis for the fluid modeling used by the code. SCRAP is a research tool, not a production code, and this report should not be considered a user's manual.

- 250 (ANL-CT--79-51) ADDED MASS AND DAMPING COEFFICIENTS FOR HEXAGONAL TUBE ARRAYS. Wilson, D.E. (Argonne National Lab., Evanston, IL (USA)). Aug 1979. Contract W-31-109-ENG-38. 24p. Dep. NTIS, PC A02/MF A01.

An analytical investigation of the fluid coupling effects from an array of hexagonal cylindrical ducts undergoing harmonic oscillations is presented. A closed form solution for the velocity and pressure is obtained under a thin gap approximation for the case of moderate frequencies. From this solution, the usual viscous and inertial fluid coupling coefficients are easily obtained. These analytically derived coefficients indicate a strong dependence upon gap spacing and oscillating Reynolds number.

- 251 (ANL-CT--80-1) COMPARATIVE TRANSIENT ANALYSIS OF METAL AND OXIDE FUELED LMFBR. Saphier, D.; Madell, J.T. (Argonne National Lab., IL (USA)). Sep 1979. Contract W-31-109-ENG-38. 125p. Dep. NTIS, PC A06/MF A01.

The neutronics and thermodynamics of an LMFBR primary system have been simulated using the DSNP simulation language. A detailed fuel pin and its subchannel model were developed and included in the DSNP library. This permits the reactor core to be simulated with any number of pins having any number of radial and axial nodes. The metal fueled core transients were compared to transients of an oxide fueled core, the conclusion being that for the same perturbations the temperature transients are faster in the metal core. A comparison between pool-type and loop-type reactors was also performed, leading to a conclusion that the transients in the upper plenum temperatures are much slower in the pool-type reactor than in the loop-type reactor.

- 252 (ANL/CT/DA--1666) EVALUATION AND INTERPRETATION OF THE ANL-MICROPHONE DATA FROM THE CAVITATION TESTS OF THE FFTF PRIMARY PUMP AT LMEC--SPTF. Cha, Y.S.; Bobis, J.P.; Carey, W.M.; Huebotter, P.R. (Argonne National Lab., IL (USA)). Jun 1977. Contract W-31-109-ENG-38. 27p. AT.

Acoustic data from the cavitation tests of the FFTF pump were analyzed. It was found that the high frequency (> 10 kHz) noise provides a

better means of detecting light and moderate pump cavitation conditions as compared with low frequency cavitation noise that was not easily distinguished from the background noise. It was observed that relatively intense cavitation occurred at an NPSH value corresponding to that when the head began to decrease. A qualitative conclusion is that, under the design conditions, short-term damage is unlikely to occur for the FFTF pump. However, the possibility of long-term (30 years) damage cannot be ruled out.

- 253 (ANL/EBR--063) DATA-PROCESSING SYSTEM FOR INSTRUMENT MAINTENANCE AND CALIBRATION AT EBR-II. Moriarty, K.J.; Ringquist, J.M. (Argonne National Lab., Idaho Falls, ID (USA); Argonne National Lab., IL (USA)). Jun 1972. Contract W-31-109-ENG-38. 47p. Dep. NTIS, PC A03/MF A01.

Early in the operation of EBR-II, it was recognized that a computer program was required for the periodic calibration and maintenance of the more than 1600 instruments in service. Computer Program No. I, set up in 1964, evolved into Computer Program No. II by addition of updated scheduling, history cards, and a new computer to execute the program. Computer Program II, combined with the required procedures, has been operative for about one year. This report describes that program and its operation.

- 254 (ANL/EBR--071) COMPUTERIZED PREVENTIVE-MAINTENANCE PROGRAM FOR EBR-II REACTOR-ASSOCIATED MECHANICAL AND POWER-ELECTRICAL COMPONENTS. Doe, C.B.; Leman, J.D.; Jacob, D.F.; Ringquist, R.M. (Argonne National Lab., Idaho Falls, ID (USA); Argonne National Lab., IL (USA)). Feb 1973. Contract W-31-109-ENG-38. 32p. Dep. NTIS, PC A03/MF A01.

This report describes the computer program that has been set up to assist in meeting the preventive-maintenance requirements of the EBR-II reactor-associated mechanical and power-electrical components. The program can provide several types of listings, including all equipment covered by the program, all equipment due for maintenance, and all equipment due for maintenance under specified plant conditions. This report also describes some other areas of the preventive-maintenance program that are not computerized. Possibilities for expansion of the computer program are briefly indicated.

- 255 (ANL/EBR--100) COOLING OF A SIMULATED EBR-II SUBASSEMBLY WITH AIR AND ARGON. Buchanan, T.N. (Argonne National Lab., IL (USA)). Jul 1978. Contract W-31-109-ENG-38. 102p. AT.

To gain more accurate knowledge of the heat-transfer characteristics of irradiated EBR-II subassemblies and to provide a broader data base for use in developing analytical models for postirradiation decay cooling, a 61-element electrically heated subassembly was constructed and instrumented so that axial and radial temperature profiles across it could be obtained. To obtain the broad data base that would be needed to fit a test to a condition that might arise, a number of tests were run under differing conditions of heat input and cooling. Cooling was done in both argon and air atmospheres and under conditions representing loss-of-flow events as well as normal cooling.

- 256 (ANL/EBR--101) COMPILED OF DATA FROM EBR-II NATURAL-CIRCULATION-TEST PROCEDURE EX-140, SECTION IV, ADDENDUM 1A. Gillette, J.L.; Sullivan, J.E.; Tokar, J.V.; Singer, R.M.

(Argonne National Lab., Idaho Falls, ID (USA)).
May 1978. Contract W-31-109-ENG-38. 257p.
AT.

Data from the XX08 natural-circulation Test 1A in EBR-II are presented herein. This test consisted of four separate coastdowns of primary-system flow initiated from approximately 5.5 percent of normal full flow (the flow provided by the auxiliary electromagnetic pump in the reactor outlet pipe of EBR-II). Each coastdown was conducted with a different flow rate of secondary sodium through the intermediate heat exchanger (IHX). Data were obtained with the EBR-II data-acquisition system (DAS) from the normal plant instrumentation, from the INCOT YY07D experiment, and from instrumented subassembly XX08, which had been installed in the reactor just before the test. The necessary corrections for offsets in the raw data collected through the DAS are discussed. The corrected data are graphically presented for each phase of the test.

257 (ANL/EBR--102) COMPILATION OF DATA FROM EBR-II NATURAL-CIRCULATION-TEST PROCEDURE EX-140, SECTION IV, ADDENDUM 1E. Gillette, J.L.; Sullivan, J.E.; Tokar, J.V.; Singer, R.M. (Argonne National Lab., IL (USA); Argonne National Lab., Idaho Falls, ID (USA)). Sep 1978. Contract W-31-7405-ENG-48. 328p. AT.

Data from the XX08 natural-circulation Test 1E are presented. This test was similar to an earlier test, XX08 Test 1A, except that instrumented subassembly XX08 had undergone irradiation before Test 1E, but not before Test 1A. Four separate coastdown transients were conducted. Each was initiated by tripping the power supply to the auxiliary electromagnetic pump, thereby causing the flow rate in the primary system to coast down to the natural-circulation flow. Each coastdown was initiated with a different flow rate in the secondary system. Data were obtained with the EBR-II data-acquisition system (DAS) from the normal plant instrumentation, from the INCOT YY07E experiment, and from instrumented subassembly XX08. The necessary corrections for offsets in the raw data collected through the DAS are discussed. The corrected data are graphically presented for each of the four phases of the test.

258 (ANL/EBR--103) RELIABILITY ANALYSIS APPLIED TO THE SRM EARTHQUAKE SHUTDOWN SYSTEM (ESS). Clay, H.W.; Lehto, W.K. (Argonne National Lab., Idaho Falls, ID (USA); Argonne National Lab., IL (USA)). Dec 1978. Contract W-31-109-ENG-38. 63p. AT.

The EBR-II Safety Research Modification (SRM) was designed to convert the EBR-II from a steady-state irradiation facility to a more versatile safety-test reactor with pulse capability. Part of this modification was to provide an upgraded shutdown system of which the ESS is a part. The fundamentals are presented of event and fault tree analysis as applied to the ESS and establishes the system reliability based on individual component failure rates and reactor-cycle operating times. As a result of this analysis, several improvements were suggested, the methodology was established for the other shutdown systems, and the reliability of the ESS was shown to be satisfactory.

259 (ANL/EBR--104) COMPILATION OF DATA FROM EBR-II NATURAL-CIRCULATION-TEST PROCEDURE EX-140, SECTION IV, ADDENDUM 7A. Gillette, J.L.; Sullivan, J.E.; Singer, R.M.; Tokar, J.V.;

Dean, E.M. (Argonne National Lab., IL (USA)). Apr 1979. Contract W-31-109-ENG-38. 167p. AT.

The transition from forced primary-coolant flow to natural-circulation flow following the loss of primary pumping power with concomitant reactor scram has been investigated in Experimental Breeder Reactor II. Before the loss-of-flow transient, the reactor was operated at 17.1 MWt (28.5% of the normal full-power value) and at 33% of normal full flow. All electrical power to the primary pumping system was terminated, causing the flow rate of the primary-system coolant to coast down. As the flow rate coasted down, the low-flow trip setting caused the reactor to scram. The transition from forced to natural-circulation flow was observed to be smooth, and the coolant temperatures throughout the reactor were observed to remain lower than the corresponding values at full-power operating conditions. The corrected test data are presented graphically.

260 (ANL/EBR--106) BIBLIOGRAPHY OF PUBLICATIONS ON EXPERIMENTAL BREEDER REACTOR NO. II (EBR-II). 1955-JULY 1979. Berg, D.B. (comp.). (Argonne National Lab., IL (USA)). Aug 1979. Contract W-31-109-ENG-38. 112p. Dep. NTIS, PC A06/MF A01.

This bibliography is divided into 15 broad areas, or categories, of interest. The same publication is listed in more than one category if its content applies to several areas of interest. Under each category, the publications are listed (1) by calendar year of publication and (2) then alphabetically by last name of the first author given. Publications that have no evident author (or compiler or editor) are listed alphabetically by title at the end of the listing for the calendar year in which they were published. Only open literature (that which can be obtained through libraries and other sources available to the public) are listed.

261 (ANL/ENG--79-05) SLSF LOOP HANDLING SYSTEM. VOLUME I. STRUCTURAL ANALYSIS. Ahmed, H.; Cowie, A.; Ma, D. (Argonne National Lab., IL (USA)). Oct 1978. Contract W-31-109-ENG-38. 453p. Dep. NTIS, PC A20/MF A01.

SLSF loop handling system was analyzed for deadweight and postulated dynamic loading conditions, identified in Chapters II and III in Volume I of this report, using a linear elastic static equivalent method of stress analysis. Stress analysis of the loop handling machine is presented in Volume I of this report. Chapter VII in Volume I of this report is a contribution by EG and G Co., who performed the work under ANL supervision.

262 (ANL/ENG--79-06) SLSF LOOP HANDLING SYSTEM. VOLUME II. CRADLE ANALYSIS FOR NORMAL AND ACCIDENTAL LOADING. Ahmed, H.; Cowie, A.; Ma, D.; Malek, R.A. (Argonne National Lab., IL (USA)). Oct 1978. Contract W-31-109-ENG-38. 400p. Dep. NTIS, PC A17/MF A01.

SLSF loop handling system was analyzed for deadweight and postulated dynamic loading conditions. Computer modeling of LHM-Cradle-transporter system is presented.

263 (ANL/ENG--79-07) SLSF LOOP HANDLING SYSTEM. VOLUME III. AISC CODE EVALUATIONS AND ANALYSIS OF CRITICAL ATTACHMENTS. Ahmed, H.; Cowie, A.; Malek, R.A.; Rafer, A.; Ma, D.; Tebo, F. (Argonne National Lab., IL (USA)). Oct 1978. Contract W-31-109-ENG-38. 263p.

- Dep. NTIS, PC A12/MF A01.
- SLSF loop handling system was analyzed for deadweight and postulated dynamic loading conditions using a linear elastic static equivalent method of stress analysis. Stress computations of Cradle and critical attachments per AISC Code guidelines are presented. HFEF is credited with in-depth review of initial phase of work.
- 264 (ANL/ENG--79-08) SLSF LOOP HANDLING SYSTEM. VOLUME IV. ANALYSIS VERIFICATIONS AND SUPPORTING APPENDICES. Ahmed, H.; Malek, R.A.; Rafer, A.; Ma, D.; D'Souza, F.; Leonard, J.W. (Argonne National Lab., IL (USA)). Oct 1978. Contract W-31-109-ENG-38. 407p. Dep. NTIS, PC A18/MF A01.
- SLSF loop handling system was analyzed for deadweight and postulated dynamic loading conditions. Some relevant backup appendices for the report are included.
- 265 (ANL/ENG--80-02) STUDY OF PIPING CONFIGURATIONS. Pan, Y.; Rafer, A.; Ahmed, H. (Argonne National Lab., IL (USA)). Jan 1980. Contract W-31-109-ENG-38. 34p. Dep. NTIS, PC A03/MF A01.
- A study of piping and elbow flexibility is performed using an analytical approach and piping analysis computer programs ADLPIPE and MARC. The study focuses on pipe loop configurations commonly used to accommodate thermal expansion in such applications as Liquid Metal Fast Breeder Reactors.
- 266 (ANL/K72--2655-1) CHARACTERIZATION OF AN ENERGY SOURCE FOR MODELING HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS IN NUCLEAR REACTORS. FIRST INTERIM REPORT. Cagliostro, D.J.; Florence, A.L. (Stanford Research Inst., Menlo Park, CA (USA)). Oct 1972. Contract W-31-109-ENG-38. 110p. Dep. NTIS, PC A06/MF A01.
- The expansion characteristics of the detonation products of a high-explosive energy source used to simulate the pressure-volume change relationships for sodium-vapor expansions during hypothetical core disruptive accidents in a Fast Test Reactor were determined experimentally. Rigid cylinder-piston experiments performed at two scales (ratio 3:3) were undertaken to determine a pressure-volume relationship as a function of source mass and expansion environment. Some of these measurements were compared with code calculations for the source.
- 267 (ANL/K76--2655-5) EXPERIMENTS ON THE RESPONSE OF RIGID AND FLEXIBLE REACTOR VESSEL MODELS TO A SIMULATED HYPOTHETICAL CORE DISRUPTIVE ACCIDENT. Cagliostro, D.J.; Romander, C.M. (Stanford Research Inst., Menlo Park, CA (USA)). Nov 1976. Contract W-31-109-ENG-38. 104p. (SRI-PYD--1966). Dep. NTIS, PC A06/MF A01.
- A simulated HCDA loading was applied in simple, thick-walled and thin-walled, 1/30-scale models of a fast breeder reactor vessel. The loading and resulting vessel response were measured to verify the modeling techniques used in the REXCO computer code. The sodium coolant is simulated by water, and the HCDA load is simulated by the expansion of the detonation products of a low-density explosive detonated in the vessel core. The detonation products expand from an initial pressure of 3700 psi down to 360 psi for a volume change of 410 cm³, equal to the cover gas volume. The expansion produces 2.87 kW-sec of gas work up to slug impact on the vessel cover. From the strain measurements and the posttest deformation profiles of the vessel, the strain energy absorbed by the vessel before and after slug impact is calculated.
- 268 (ANL/MSD--78-3) COMPILATION OF FATIGUE DATA FOR INCOLOY 800 ALLOYS. Majumdar, S. (Argonne National Lab., IL (USA)). Mar 1978. Contract W-31-109-ENG-38. 34p. Dep. NTIS, PC A03/MF A01.
- The present report is a compilation of all room- and elevated-temperature, low-cycle fatigue data for Incoloy alloys 800 and 800H that are available in the literature in the United States. The compilation represents 482 data points that include strain-controlled, stress-controlled, and rotating-beam tests.
- 269 (ANL-RDP--61) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MAY--JUNE 1977. (Argonne National Lab., IL (USA)). 11 Aug 1977. Contract W-31-109-ENG-38. vp. AT.
- Progress is summarized in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), LMFBR Design Support, Instrumentation and Control, Fuels and Materials Development, Reactor Physics, Reactor Safety, and Gas Cooled Fast Breeder Reactors.
- 270 (ANL-RDP--62) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, JULY--AUGUST 1977. Sachs, R.G.; Kyger, J.A. (Argonne National Lab., IL (USA)). 5 Oct 1977. Contract W-31-109-ENG-38. vp. AT.
- Progress is summarized in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), LMFBR Design Support, Instrumentation and Control, Reactor Physics, Reactor Safety, and Other Fast Breeder Reactors.
- 271 (ANL-RDP--63) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT, SEPTEMBER 1977. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 170p. AT.
- This progress report summarizes work performed during September 1977 in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), LMFBR Design Support, Instrumentation and Control, Reactor Physics, and Reactor Safety.
- 272 (ANL-RDP--64) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT. (Argonne National Lab., IL (USA)). Oct 1977. Contract W-31-109-ENG-38. 169p. AT.
- The report summarizes work performed in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), LMFBR design support, instrumentation and control, reactor physics, and reactor safety.
- 273 (ANL-RDP--65) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT, NOVEMBER 1977. Sachs, R.G.; Kyger, J.A. (Argonne National Lab., IL (USA)). 24 Jan 1978. Contract W-31-109-ENG-38. 166p. AT.
- Progress is summarized on work performed during November 1977 in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), LMFBR design support, instrumentation and control, reactor physics, reactor safety, and GCFR type reactors.
- 274 (ANL-RDP--66) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, DECEMBER 1977. Sachs, R.G.; Kyger, J.S. (Argonne National

Lab., IL (USA)). 17 Feb 1978. Contract W-31-109-ENG-38. 163p. AT.

Progress is summarized on work performed during December 1977 in the following research and development areas: experimental breeder reactor No. II (EBR-II), breeder-reactor design support, instrumentation and control, reactor physics, reactor safety, and other fast breeder reactors.

- 275 (ANL-RDP--67) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, JAN 1978. Sachs, R.G.; Kyger, J.A. (Argonne National Lab., IL (USA)). Jan 1978. Contract W-31-109-ENG-38. 100p. AT.

This progress report summarizes the Argonne National Laboratory work performed for the Reactor Development Program during January 1978 in the following research and development areas: Experimental Breeder Reactor No. II, Breeder-reactor Design Support, Instrumentation and Control, Fuels and Materials Development, Reactor Physics, Reactor Safety, and Gas Cooled Fast Breeder Reactors.

- 276 (ANL-RDP--68) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, FEBRUARY 1978. (Argonne National Lab., IL (USA)). 7 Apr 1978. Contract W-31-109-ENG-38. 173p. AT.

Progress is summarized in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), Breeder-reactor Design Support, Instrumentation and Control, Reactor Physics, Reactor Safety, and Other Fast Breeder Reactors.

- 277 (ANL-RDP--69) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, MARCH 1978. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 156p. AT.

This progress report summarizes the Argonne National Laboratory work performed for the Reactor Development Program during March 1978 in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), Breeder-reactor Design Support, Instrumentation and Control, Fuels and Materials Development, Reactor Physics, Reactor Safety, and Other Fast Breeder Reactors.

- 278 (ANL-RDP--70) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT, APRIL 1978. (Argonne National Lab., IL (USA)). 16 Jun 1978. Contract W-31-109-ENG-38. 186p. AT.

The report summarizes work in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), Breeder-reactor Design Support, Instrumentation and Control, Reactor Physics, Reactor Safety, and GCFR Type Reactors.

- 279 (ANL-RDP--71) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT, MAY 1978. Sachs, R.G.; Kyger, J.A. (Argonne National Lab., IL (USA)). 12 Jul 1978. Contract W-31-109-ENG-38. 165p. AT.

Progress is summarized in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), breeder-reactor design support, instrumentation and control, reactor physics, reactor safety, energy and environmental systems, and gas-cooled fast breeder reactors.

- 280 (ANL-RDP--72) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT, JUNE 1978. (Argonne National Lab., IL (USA)). 11 Aug 1978. Contract W-31-109-ENG-38. 200p. AT.

The report summarizes work performed in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), Breeder-Reactor Design Support, Instrumentation and Control, Reactor Physics, Reactor Safety, and Gas Cooled Fast Breeder Reactors.

- 281 (ANL-RDP--73) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, JULY 1978. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 119p. AT.

Progress is summarized in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), Breeder-reactor Design Support, Instrumentation and Control, Reactor Physics, Reactor Safety, and Gas Cooled Fast Breeder Reactors.

- 282 (ANL-RDP--74) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, AUGUST 1978. (Argonne National Lab., IL (USA)). 18 Oct 1978. Contract W-31-109-ENG-38. 150p. AT.

Progress is summarized in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), Breeder-reactor Design Support, Instrumentation and Control, Reactor Physics, Reactor Safety, and Gas Cooled Fast Breeder Reactors.

- 283 (ANL-RDP--75) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT. (Argonne National Lab., IL (USA)). Sep 1978. Contract W-31-109-ENG-38. 181p. AT.

Progress is summarized in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), Breeder-reactor Design Support, Instrumentation and Control, Reactor Physics, Reactor Safety, and Gas Cooled Fast Breeder Reactors.

- 284 (ANL-RDP--76) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT, OCTOBER 1978. Sachs, R.G.; Kyger, J.A. (Argonne National Lab., IL (USA)). 15 Dec 1978. Contract W-31-109-ENG-38. 131p. AT.

Progress is summarized in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), Breeder-reactor Design Support, Instrumentation and Control, Fuels and Materials Development, Reactor Physics, and Reactor Safety.

- 285 (ANL-RDP--77) REACTOR DEVELOPMENT PROGRAM PROGRESS REPORT, NOVEMBER 1978. Sachs, R.G.; Kyger, J.A. (Argonne National Lab., IL (USA)). 16 Jan 1979. Contract W-31-109-ENG-38. 116p. AT.

Progress is summarized in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), breeder-reactor design support, instrumentation and control, reactor physics, and reactor safety.

- 286 (ANL-RDP--78) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT, DECEMBER 1978. (Argonne National Lab., IL (USA)). 20 Feb 1979. Contract W-31-109-ENG-38. 129p. AT.

Progress is summarized in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II), Breeder Reactor Design Support, Instrumentation and Control, Reactor Physics, Reactor Safety, and Other Fast Breeder Reactors.

- 287 (ANL-RDP--79) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT. (Argonne National Lab., IL (USA)). Jan 1979. Contract W-31-

109-ENG-38. 136p. AT.

Progress is summarized in the following research and development areas: Experimental Breeder Reactor No. II (EBR-II) breeder-reactor design support, engineered systems and components, reactor physics, reactor safety, and other fast breeder reactors.

288 (ANL-RDP--80) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT, FEBRUARY 1979. (Argonne National Lab., IL (USA)). 19 Apr 1979. Contract W-31-109-ENG-38. 116p. AT.

Progress is summarized in the following research and development areas: Safety, Engineered Systems and Components, Physics, Gas Cooled Breeder Reactor, and Test Facilities.

289 (ANL-RDP--81) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT, MARCH 1979. (Argonne National Lab., IL (USA)). 16 May 1979. Contract W-31-109-ENG-38. 121p. AT.

This progress report summarizes the Argonne National Laboratory work performed for the Reactor Development Program during March 1979 in the following research and development areas: Safety, Engineered Systems and Components, Physics, Gas Cooled Breeder Reactor, Test Facilities - ANL, and EBR-II Fuel Fabrication Costs Inventory Account.

290 (ANL-RDP--82) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT. (Argonne National Lab., IL (USA)). Apr 1979. Contract W-31-109-ENG-38. 181p. AT.

This progress report summarizes the Argonne National Laboratory work performed for the Reactor Development Program during April 1979 in the following research and development areas: safety, engineered systems and components, physics, gas cooled breeder reactor, and test facilities - ANL.

291 (ANL-RDP--83) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT. (Argonne National Lab., IL (USA)). May 1979. Contract W-31-109-ENG-38. 119p. AT.

This progress report summarizes the Argonne National Laboratory work performed for the Reactor Development Program during May 1979 in the following research and development areas: safety, physics, fuel and absorber development, test facilities - ANL, and EBR-II fuel fabrication costs inventory account.

292 (ANL-RDP--84) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT. (Argonne National Lab., IL (USA)). Jun 1979. Contract W-31-109-ENG-38. 182p. AT.

This progress report summarizes the Argonne National Laboratory work performed for the Reactor Development Program during June 1979 in the following research and development areas: Safety, Engineered Systems and Components, Physics, Gas Cooled Breeder Reactor, and Test Facilities - ANL.

293 (ANL-RDP--85) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT, JULY 1979. (Argonne National Lab., IL (USA)). Jul 1979. Contract W-31-109-ENG-38. 109p. AT.

This progress report summarizes the Argonne National Laboratory work performed for the Reactor Development Program during July 1979 in the following research and development areas: Safety, Engineered Systems and Components, Physics, Fuel and Absorber Development, Test Facilities - ANL, and EBR-II Fuel Fabrication Costs Inventory Account.

294 (ANL-RDP--86) REACTOR DEVELOPMENT PROGRAM. PROGRESS REPORT, AUGUST 1979. Massey, W.E.; Kyger, J.A. (Argonne National Lab., IL (USA)). 15 Oct 1979. Contract W-31-109-ENG-38. 86p. AT.

Progress is summarized in the following research and development areas: Safety, Physics, Gas Cooled Breeder Reactor, and Test Facilities.

295 (AWRE--49/91/87) COMPUTER CODE FOR SEURBNUK-2 FOR FAST REACTOR EXPLOSION CONTAINMENT SAFETY STUDIES. Cameron, I.G.; Hankin, B.C.; Warham, A.G.P.; Benuzzi, A.; Yerkes, A. (UKAEA Weapons Group, Aldermaston. Atomic Weapons Research Establishment; Commission of the European Communities, Ispra (Italy). Joint Research Centre). May 1977. 26p. (TRG-Report--3006(R/X); JRC-ISPRA-ET--02/77). Dep. NTIS (US Sales Only), PC A03/MF A01.

The SEURBNUK-2 code has been developed jointly by JRC, Ispra and AWRE, Aldermaston for use in containment analysis for fast reactor safety studies. It is a major extension of the SEURBNUK code developed earlier by Belgonucleaire and Aldermaston, which, in turn, was based on the Marker and Cell (MAC) code. In the Marker and Cell method the motion of a two-dimensional incompressible fluid, which may have a free surface, is solved using an Eulerian approach whereby the fluid moves through a fixed grid of mesh lines with velocity and pressure as the basic variables. Massless particles are distributed throughout the fluid and move with it, but their sole use is to determine which cells contain fluid. The numerical procedures adopted in SEURBNUK to solve the basic hydrodynamic equations are similar to the ICE procedure which itself is an extension of the MAC method. In this the non-linear partial differential equations are solved by an implicit finite difference technique which advances the solution in a series of time steps. SEURBNUK-2 has a full thin shell treatment for tanks of arbitrary shape and includes the effects of compressibility of the fluid. A feature of SEURBNUK is that the equations of motion for the thin shells are solved quite separately from those for the fluid and the time step for the fluid flow calculation is an integral multiple of that for calculating the shell motion. The interaction of the shell with the fluid is then considered as a modification to the coefficients in the implicit pressure equations, the modification naturally depending on the behavior of the thin shell section within the fluid cell.

296 (BLG--515) ANNUAL SCIENTIFIC REPORT 1975. Billiau, R.; Bobin, K.; Michiels, G.; Proost, J. (eds.). (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)). 1976. 236p. Dep. NTIS (US Sales Only), PC A02/MF A01.

The main activities of SCK/CEN during 1975 are reported in individual summaries. Field of research are the following: sodium cooled fast reactors, gas cooled reactors, light water reactors, applied nuclear research (including waste disposal, safeguards and fusion research), basic and exploratory research (including materials science, nuclear physics and radiobiology). The BR2 Materials testing reactor and associated facilities are described. The technical and administrative support activities are also presented. A list of publications issued by the SCK/CEN Scientific staff is given.

- 297 (BLG--520) ANNUAL SCIENTIFIC REPORT 1976. Billiau, R.; Kirk, F.; Proost, J. (eds.). (Centre d'Etude de l'Energie Nucleaire, Mcl (Belgium)). 1977. vp. Dep. NTIS (US Sales Only), PC A09/MF A01.

This report of the Centre d'Etude de l'Energie Nucleaire - Studiecentrum voor Kernenergie gives a survey of the scientific and technical work done in 1976. The research areas were as follows: 1. The sodium-cooled fast reactors and, in particular, mixed oxide fuels, carbide fuel, materials development, reprocessing, fast reactor physics, safety and instrumentation, and sodium technology. 2. The gas-cooled reactors as gas-cooled fast and high-temperature reactors. 3. The light water reactors, in particular, the BR3 reactor, light-water reactor fuels and plutonium recycling. 4. Applied nuclear research, waste processing and disposal: safeguards, fusion research and lithium technology. 5. Basic and exploratory research in materials science and nuclear physics. 6. Non-nuclear research and development such as in air pollution, pollution abatement and waste handling, fuel cells and applied electrochemistry.

- 298 (BLG--524) ANNUAL SCIENTIFIC REPORT 1977. Proost, J.; Billiau, R.; Kirk, F. (eds.). (Centre d'Etude de l'Energie Nucleaire, Mcl (Belgium)). 1978. vp. Dep. NTIS (US Sales Only), PC A09/MF A01.

This report of the Centre d'Etude de l'Energie Nucleaire - Studiecentrum voor Kernenergie gives a survey of the scientific and technical work done in 1977. The research areas are: 1. The sodium cooled fast reactors and namely the mixed oxide fuels, the carbide fuel, the materials development, the reprocessing, the fast reactor physics, the safety and instrumentation and the sodium technology. 2. The gas cooled reactors as gas cooled fast and high temperature reactors. 3. The light water reactors namely the BR3 reactor, the light water reactor fuels and the plutonium recycling. 4. The applied nuclear research, waste conditioning and disposal as the safeguards, the fusion research and the lithium technology. 5. The basic and exploratory research namely the materials science and the nuclear physics and finally 6. Non-nuclear research and development such as the air pollution, the pollution abatement and waste handling, the fuel cells and applied electrochemistry.

- 299 (BMI-NUREG--1983) AEROSOL MEASUREMENTS AND MODELING FOR FAST REACTOR SAFETY.

QUARTERLY PROGRESS REPORT, APRIL 1--JUNE 30, 1977. TASK 7. Gieseke, J.A.; Jordan, H.; Schmidt, E.W.; Lee, K.W.; Vaishnavi, B.; Reed, L.D. (Battelle Columbus Labs., OH (USA)). 28 Oct 1977. 24p. NTIS, PC A03/MF A01.

Technical efforts were concentrated on improving the HAARM-2 code primarily by adding the results of the sodium oxide agglomerate measurements, on formulating the mathematical basis for reference code development, on investigating the possibilities for using polynomials to express particle size distributions in the reference code, and on evaluating the use of a laser Doppler interferometry system to measure sodium oxide agglomerate sizes. Details of efforts performed and progress made on tasks in these several general areas are discussed.

- 300 (BMI-NUREG--1987) AEROSOL MEASUREMENTS AND MODELING FOR FAST REACTOR SAFETY: TASK 7.

ANNUAL REPORT FOR FY 1977. Gieseke, J.A.; Jordan, H.; Lee, K.W.; Vaishnavi, B.; Reed, L.D. (Battelle Columbus Labs., OH (USA)). 23 Dec 1977. 114p. Dep. NTIS, PC A06/MF A03.

Progress is summarized in the following areas: (1) measurements of agglomerate characteristics and thermophoretic forces for sodium oxide aerosols, (2) revision of HAARM-2 computer code, (3) comparison of HAARM-2 and HAARM-3 calculations, (4) evaluation of BAM code capabilities, (5) development of a reference code, and (6) experiments to characterize fuel material aerosol properties. A compilation of experimental data is presented. (DG)

- 301 (BMI-NUREG--1991) HAARM-3 USERS MANUAL.

Gieseke, J.A.; Lee, K.W.; Reed, L.D. (Battelle Columbus Labs., OH (USA)). 5 Jan 1978. 73p. NTIS, PC A04/MF A01.

Various models have been developed to describe aerosol behavior under postulated LMFBR accident conditions. Each of the codes has its own advantages and limitations regarding the accuracy of the physically realistic model of the actual aerosol behavior. The present computer code designated HAARM-3 (Heterogeneous Aerosol Agglomeration Revised Model-3) is a new version of the previous HAARM-2 and HAARM-1 models. One of the main advantages of these codes is their relatively short computing times. Therefore, these codes have been widely used for nuclear reactor safety analyses. The numerical methods employed in the present codes are based on the previous HAA-3B code which was originally developed at Atomics International. By assuming that the time dependent particle size distribution is lognormal, the integro-differential equation describing the concentration decay of aerosol is converted into three first-order differential equations. These equations with several additional first order differential equations for the generation of other desired information are integrated using a standard library subroutine. The model described in the report differs from the previous models in that all of the aerosol agglomeration and deposition mechanisms which are dependent upon the morphological properties of agglomerates are corrected using a more recent experimental density correction factor. The effects of non-spherical shape of the agglomerates on settling velocity and collision rate may also be accounted for at the user's discretion. In addition, aerosol agglomeration due to turbulence which was not included before has been added.

- 302 (BN--7612-04) RESULTS OF THE POST-IRRADIATION EXAMINATIONS OF THE RAPSODIE I EXPERIMENT FUEL PINS. Haas, V.; Gaube, M.; van de Velde, J.; Ketels, J.; van Loon, C. (Societe Belge pour l'Industrie Nucleaire, Brussels). 1976. 25p. Dep. NTIS (US Sales Only), PC A02/MF A01.

An experiment sponsored by GfK, BELGONUCLEAIRE and INTERATOM has been carried out in the French reactor Rapsodie from February 1971 to September 1972, in the frame of the R and D program for the fast breeder prototype reactor SNR. This experiment consisted in the irradiation of two bundles of 34 mixed oxide pins each, with spacer grids and tie-rods, up to a peak burn-up of 19.6 a/o and a peak total neutron fluence of $8 \times 10^{20} \text{ mm}^{-2}$, corresponding to a peak fast fluence ($E > 0.111 \text{ MeV}$) of $6.4 \times 10^{20} \text{ mm}^{-2}$. The post-irradiation examinations showed that the cold-worked WNr. 1.4970 cladding stainless steel has a lower irradiation induced swelling than the solution-

annealed WNr. 1.4988. Important inelastic deformations of both types of claddings have been observed; the largest part of these deformations can be explained by the in-pile creep. A mechanical interaction between the fuel and the cladding is not excluded in the case of the pins with the cladding steel WNr. 1.4970. The destructive examinations emphasized the influence of the pellets initial stoichiometry on the fission products migration, on the plutonium radial distribution and on the fuel-cladding chemical interaction. An interaction between the fuel and the blanket pellets due to the cesium migration and to the formation of cesium uranates, leading to local cladding strains, has been noticed. These phenomena, as well as the chemical interaction between the fuel and the cladding, have been investigated by extensive microprobe analysis. Finally, the fuel restructuration has been analysed and interpreted as a function of the fuel characteristics, irradiation conditions and cladding deformations.

- 303 (BNL--50831-4) DESIGN GUIDE FOR CATEGORY IV REACTORS: LIQUID METAL REACTORS. Brynda, W.J.; Karol, R.C.; Lobner, P.R.; Powell, R.W.; Straker, E.A. (Brookhaven National Lab., Upton, NY (USA)). Jul 1979. Contract EY-76-C-02-0016. 243p. Dep. NTIS, PC A11/MF A01.

The purpose of this Design Guide is to provide additional guidance to aid the DOE facility contractor in meeting the requirement that the siting, design, construction, modification, operation, maintenance, and decommissioning of DOE-owned reactors be in accordance with generally uniform standards, guides, and codes which are comparable to those applied to similar reactors licensed by the Nuclear Regulatory Commission (NRC). This Design Guide deals principally with the design and functional requirements of liquid metal cooled fast reactor (Category IV reactor) structures, components, and systems.

- 304 (BNL--50992, pp 287-295) REACTOR APPLICATION OF ACTINIDE CROSS SECTION DATA. Bearan, S.L.; Cowan, C.L. (General Electric Co., Sunnyvale, CA). May 1979.

From Conference on the nuclear data of higher plutonium and americium isotopes for reactor applications; Upton, NY, USA (20 Nov 1978).

Uncertainties in the cross section data for the higher plutonium and americium isotopes have been investigated to determine their impact on the important LMFBR design parameters. The results of this investigation have indicated that the design calculations are sensitive to the uncertainties in the capture and fission data for ^{240}Pu and ^{241}Pu , and for long-term design applications these uncertainties should be reduced to less than about 10%. The uncertainties in the data for ^{242}Pu and for the americium isotopes have only a small impact on the fast reactor design studies but may be important in the development of schemes for the management of actinide waste discharge and for plutonium burnout.

- 305 (BNL-NUREG--23113) SIMULATION OF OPERATIONAL AND SAFETY TRANSIENTS IN LMFBR SYSTEMS. Agrawal, A.K.; Guppy, J.G. (Brookhaven National Lab., Upton, NY (USA)). 1978. Contract EY-76-C-02-0016. 6p. (CONF-780407--1). Dep. NTIS, PC A02/MF A01.

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

A new, advanced thermohydraulic simulation code for LMFBRs is described. This code, designated as the Super System Code (SSC), simulates operational and safety transients that may be initiated anywhere in the system. The SSC code provides for the simulation of all essential components in the primary and intermediate heat transport systems. The steam generating circuit along with the turbine and condenser are also included. The usefulness of this code in future LMFBR plant designs is demonstrated by its application to the Clinch River Breeder Reactor-type plant. Several design and operational transients such as normal scram, control rod withdrawal, after-heat removal by natural convection and load follow-up are analyzed. A pipe rupture accident in the primary heat transport system is also discussed.

- 306 (BNL-NUREG--24043) REVIEW OF THE FFTF NATURAL CIRCULATION TEST PLANS. Perkins, K.R.; Bari, R.A.; Chen, L. (Brookhaven National Lab., Upton, NY (USA)). Feb 1978. Contract EY-76-C-02-0016. 47p. Dep. NTIS, PC A03/MF A01.

A review of the FFTF Natural Circulation test plans has been conducted in order to evaluate their relationship to the safety of the facility and to verification of analytical methods. It is concluded that: in general, the bases for the test acceptance criteria are incomplete and do not achieve the desired objective of assuring adequate natural circulation decay heat removal for "worst case" conditions; and the behavioral gap between the 5 percent and 75 percent power transients is too wide to assure the safety of the facility (without extensive uncertainty analyses which apparently have not been performed). Recommendations are presented.

- 307 (BNL-NUREG--24141-R) CONTAINMENT RESPONSE TO POSTULATED CORE MELTDOWN ACCIDENTS IN THE FAST FLUX TEST FACILITY. Gasser, R.D.; Pratt, W.T. (Brookhaven National Lab., Upton, NY (USA)). Aug 1978. Contract EY-76-C-02-0016. 100p. Dep. NTIS, PC A05/MF A01.

An assessment of the containment margin available in the Fast Flux Test Facility to mitigate the consequences of a postulated failure on in-vessel post-accident heat removal following a hypothetical core-disruptive accident (HCDA) has been completed. Two general meltdown configurations (termed in-vessel and ex-vessel) have been considered, and it is concluded that: (1) For the in-vessel meltdown scenario, the quantity of sodium assumed ejected from the vessel into the cavity during the HCDA is crucial to the subsequent reactor containment building (RCB) pressurization. If the reactor cavity liner is also assumed to fail, then the containment integrity could be challenged within 10 hours. (2) For the ex-vessel meltdown scenario, the collapse of the reactor cavity floor, which allows sodium to contact the unlined subcavity, is crucial to the subsequent RCB pressurization. Predictions of cavity floor penetration vary between 2 to 60 hours depending on the initial assumptions made; however, after floor collapse occurs overpressurization of the RCB occurs rapidly within several hours.

- 308 (BNL-NUREG--24162) RELIABILITY OF THE CRBRP SHUTDOWN HEAT REMOVAL SYSTEM. Buslik, A.J.; Papazoglou, I.A.; Bari, R.A. (Brookhaven National Lab., Upton, NY (USA); Massachusetts Inst. of Tech., Cambridge (USA)). 1978. Contract EY-76-C-02-0016. 12p. (CONF-780507--13). Dep. NTIS, PC A02/MF A01.

From Probabilistic analysis of nuclear reactor safety; Los Angeles, CA, USA (8 May 1978).

The failure probability per year of the shutdown heat removal system (SHRS) of the Clinch River Breeder Reactor Plant is calculated for the loss of off-site power (LOSP) initiator. Only the case in which reactor operation is not permitted on less than 3 loops is considered. The design studied includes the capability of operating pony motors on the pumps in one primary and associated intermediate loop from a battery, and includes a Direct Heat Removal Service which under certain circumstances can lead to successful decay heat removal even if heat transport through the main heat transport system is cut off at the intermediate heat exchangers. The calculated failure probability per year of the SHRS due to LOSP ranges from $3 \times 10^{-6}/\text{yr}$ to $4 \times 10^{-6}/\text{yr}$, depending on the diesel generator failure probabilities used, the assumptions concerning adequacy of natural circulation in main heat transport system sodium loops, the requirements for auxiliary feedwater system mission success, and the probability that off-site power is not restored in two hours.

- 309 (BNL-NUREG--24946) ASSESSMENT OF THE LOSS-OF-HEAT-SINK ACCIDENT WITH SCRAM IN THE LMFBR. Bari, R.A.; Ludewig, H.; Pratt, W.T.; Sun, Y.H. (Brookhaven National Lab., Upton, NY (USA)). 1978. Contract EY-76-C-02-0016. 12p. (CONF-781022--37). Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

A description of a slow core meltdown in a liquid metal fast breeder reactor is presented for the conditions of loss-of-heat-sink following neutronic shutdown. Simple models are developed for the prediction of phase changes and/or relocation of the core materials including fuel, clad, ducts, control rod absorber material (B₄C), and plenum gases. The sequence of events is accounted for and the accident progression is described up to the point of recriticality. The neutronic behavior of the disrupted core is analyzed in R-Z geometry with a static transport theory code. For most scenarios assessed, the reactor is expected to become recritical although large ramp rates are not anticipated.

- 310 (BNL-NUREG--24984) VOLUME-HEATED BOILING POOL FLOW BEHAVIOR AND APPLICATION TO TRANSITION PHASE ACCIDENT CONDITIONS. Ginsberg, T.; Jones, D.C. Jr.; Chen, J.C. (Brookhaven National Lab., Upton, NY (USA)); Lehigh Univ., Bethlehem, PA (USA). Dept. of Mechanical Engineering and Mechanics). 1978. Contract EY-76-C-02-0016. 20p. (CONF-781022--39). Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

Observations of two-phase flow fields in volume-heated boiling pools are reported. Photographic observations, together with pool-average void fraction measurements are presented. Flow regime transition criteria derived from the measurements are discussed. The churn-turbulent flow regime was the dominant regime for superficial vapor velocities up to nearly five times the Kutateladze dispersal velocity. Within this range of conditions, a churn-turbulent drift flux model provides a reasonable prediction of the pool-average void fraction data. The results of the experiment and analyses are extrapolated to transition phase conditions. It is shown that intense pool boil-up could occur

where the pool-average void fraction would be greater than 0.6 for steel vaporization rates equivalent to power levels greater than one percent of nominal LMFBR power density.

- 311 (BNL-NUREG--24988) CONFIRMATORY SIMULATION OF SAFETY AND OPERATIONAL TRANSIENTS IN LMFBR SYSTEMS. Guppy, J.G.; Agrawal, A.K. (Brookhaven National Lab., Upton, NY (USA)). 1978. Contract EY-76-C-02-0016. 12p. (CONF-781022--40). Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

Operational and safety transients (anticipated, unlikely, or extremely unlikely) that may originate anywhere in a liquid-metal fast breeder reactor (LMFBR) system must be adequately simulated to assist in safety evaluation and plant design efforts. An advanced thermohydraulic transient code, the Super System Code (SSC), is described that may be used for confirmatory safety evaluations of plant-wide events, such as assurance of adequate decay heat removal capability under natural circulation conditions. Results obtained with SSC illustrating the degree of modeling detail present in the code as well as the computing efficiency are presented. A version of the SSC code, SSC-L, applicable to any loop-type LMFBR design, has been developed at Brookhaven. The scope of SSC-L is to enable the simulation of all plant-wide transients covered by Plant Protection System (PPS) action, including sodium pipe rupture and coastdown to natural circulation conditions. The computations are stopped when loss of core integrity (i.e., clad melting temperature exceeded) is indicated.

- 312 (BNL-NUREG--25140) STATE-OF-THE-ART REVIEW OF SODIUM FIRE ANALYSIS AND CURRENT NOTIONS FOR IMPROVEMENTS. Tasi, S.S. (Brookhaven National Lab., Upton, NY (USA)). 1978. Contract EY-76-C-02-0016. 13p. (CONF-781173--1). Dep. NTIS, PC A02/MF A01.

From Specialists' meeting on sodium fires and prevention; Aix-en-Provence, France (20 Nov 1978).

Sodium releases from postulated pipe ruptures, as well as failures of sodium handling equipment in liquid metal fast breeder reactors, may lead to substantial pressure-temperature transients in the sodium system cells, as well as in the reactor containment building. Sodium fire analyses are currently performed with analytical tools, such as the SPRAY, SOMIX, SPOOL-FIRE and SOFIRE-II codes. A review and evaluation of the state-of-the-art in sodium fire analysis is presented, and suggestions for further improvements are made. This work is based, in part, on studies made at Brookhaven National Laboratory during the past several years in the areas of model development and improvement associated with the accident analyses of LMFBRs.

- 313 (BNL-NUREG--25297) FLOW DYNAMICS OF VOLUME-HEATED BOILING POOLS. Ginsberg, T.; Jones, D.C.; Chen, J.C. (Brookhaven National Lab., Upton, NY (USA)); Lehigh Univ., Bethlehem, PA (USA). Dept. of Mechanical Engineering and Mechanics). 1979. Contract EY-76-C-02-0016. 13p. (CONF-790808--2). Dep. NTIS, PC A02/MF A01.

From 18. ASME national heat transfer conference; San Diego, CA, USA (Aug 1979).

Safety analyses of fast breeder reactors require understanding of the two-phase fluid dynamic and heat transfer characteristics of volume-heated boiling pool systems. Design of direct contact three-phase boilers, of

practical interest in the chemical industries also requires understanding of the fundamental two-phase flow and heat transfer behavior of volume boiling systems. Several experiments have been recently reported relevant to the boundary heat-loss mechanisms of boiling pool systems. Considerably less is known about the two-phase fluid dynamic behavior of such systems. This paper describes an experimental investigation of the steady-state flow dynamics of volume-heated boiling pool systems.

- 314 (BNL-NUREG--25421) APPLICABILITY OF MICROWAVE HEATING TO THE SIMULATION OF THE LMFBR MELTDOWN PROCESS. Makowitz, H.; Ginsberg, T. (Brookhaven National Lab., Upton, NY (USA)). Dec 1978. Contract EY-76-C-02-0016. 7p. (CONF-790602--11). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

In the analysis of post accident heat removal (PAHR) following a hypothetical core-disruptive accident (HCDA) in an LMFBR one is faced with the problem of simulating a boiling pool of a fissile fuel, steel mixture. Previous work at Argonne National Laboratory has recommended that a small scale simulation study of melt-through by microwave heating would be useful on the basis of scoping studies which indicate that the internal heating is quite uniform. In the present work, the effects of droplet size, temperature, dielectric properties and wavelength on the absorption efficiency are investigated and the appropriate materials for obtaining near equal volume heating over a range of droplet sizes of 5.0 x 10⁻⁴ cm to 10 cm in radius are suggested for a suitable wavelength.

- 315 (BNL-NUREG--25492) COMBUSTION OF LARGE SODIUM POOLS. Tsai, S.S.; Bari, R.A. (Brookhaven National Lab., Upton, NY (USA)). 1979. Contract EY-76-C-02-0016. 11p. (CONF-790816--7). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Concern over potential spillages of sodium in liquid metal fast breeder reactors has prompted several experimental and theoretical investigations of sodium pool fires in recent years. Newman has measured the burning rates of sodium pool fires and proposed a correlation for the burning rates based on an analogy between mass transfer and heat transfer. In his theory, only oxygen transport to the pool surface is considered and, therefore, experimental data on the flame temperature is needed to compute the pool burning rate. The determination of the flame structure and the pool burning rate are analyzed. The present theory considers not only the oxygen transport to the flame but also the thermal interactions between the flame and the pool surface and between the flame and the ambient. This makes the determination of the flame temperature and position possible.

- 316 (BNL-NUREG--25493) ANALYSIS OF THE FFTF PRIMARY PIPE RUPTURE TRANSIENTS. Perkins, K.R.; Bari, R.A.; Chen, L.C.; Albright, D.C. (Brookhaven National Lab., Upton, NY (USA)). 1979. Contract EY-76-C-02-0016. 26p. (CONF-790816--6). Dep. NTIS, PC A03/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The response of the Fast Flux Test Facility (FFTF) to hypothetical ruptures of the high pressure primary piping has been analyzed using

two LMFBR plant systems codes, namely IANUS and DEMO. Comparisons of the average channel temperatures predicted by the two codes show good agreement for identical transients. However, the hot channel temperatures predicted by DEMO are about 60K higher than the corresponding IANUS predictions for severe transients. This difference is attributed to the dynamic hot channel factors employed in DEMO which discount the thermal inertia of the duct walls for rapid transients. DEMO also predicts more severe transients for hot-leg ruptures in FFTF than previously reported analyses for the CRBR.

- 317 (BNL-NUREG--25497) FEEDBACK CONTROL SYSTEMS FOR NON-LINEAR SIMULATION OF OPERATIONAL TRANSIENTS IN LMFBRs. Khatib-Rahbar, M.; Agrawal, A.K.; Srinivasan, E.S. (Brookhaven National Lab., Upton, NY (USA)). [nd]. Contract EY-76-C-02-0016. 9p. Dep. NTIS, PC A02/MF A01.

Adequate modeling of Plant Control Systems (PCS) for the study of Anticipated Transients Without Scram (ATWS) is of considerable significance in the design, operation and safety evaluation of Liquid-Metal-Cooled Fast Breeder Reactor (LMFBR) systems. To assess the system response to high frequency, low consequence events, the plant needs to be dynamically simulated. The description of analytical and numerical models for PCS that have been developed and incorporated into the loop version of the Super System Code (SSC=L) are described. The importance of detailed modeling of control systems is discussed. Sample transient results obtained for a 20% ramp change of load in 40 s in the Clinch River Breeder Reactor Plant (CRBRP) are also shown.

- 318 (BNL-NUREG--25498) SINGLE-PHASE SODIUM PUMP MODEL FOR LMFBR THERMAL-HYDRAULIC ANALYSIS. Madni, I.K.; Cazzoli, E.G.; Agrawal, A.K. (Brookhaven National Lab., Upton, NY (USA)). 1979. Contract EY-76-C-02-0016. 28p. (CONF-790816--10). Dep. NTIS, PC A03/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

A single-phase, homologous pump model has been developed for simulation of safety-related transients in LMFBR systems. Pump characteristics are modeled by homologous head and torque relations encompassing all regimes of operation. These relations were derived from independent model test results with a centrifugal pump of specific speed equal to 35 (SI units) or 1800 (gpm units), and are used to analyze the steady-state and transient behavior of sodium pumps in a number of LMFBR plants. Characteristic coefficients for the polynomials in all operational regimes are provided in a tabular form. The speed and flow dependence of head is included through solutions of the impeller and coolant dynamic equations. Results show the model to yield excellent agreement with experimental data in sodium for the FFTF prototype pump, and with vendor calculations for the CRBR pump. A sample pipe rupture calculation is also performed to demonstrate the necessity for modeling the complete pump characteristics.

- 319 (BNL-NUREG--25506) SURVEY AND EVALUATION OF FFTF INSTRUMENTATION FOR SYSTEMS CODE VALIDATION. Majumdar, D.; Ludwig, H.; Cerbone, R.J. (Brookhaven National Lab., Upton, NY (USA)). 1979. Contract EY-76-C-02-0016. 14p. (CONF-790816--9). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

FFTF instrumentation and preoperational tests are examined here to determine whether the proposed tests and associated instrumentation can provide relevant information needed for verification of systems codes such as SSC. Details of both the in-core and out-of-core instrumentations are studied and their accuracy examined. From a series of multichannel core calculations, the temperatures in instrumented subassemblies were correlated with temperatures in uninstrumented subassemblies. It is found that FFTF can furnish relevant steady-state data at different core conditions for use in systems code validation. The two FOTA channels will provide the most valuable in-core data. However, the FFTF natural circulation tests should be considered as only one of a series of experiments necessary to validate a large systems code.

- 320 (BNL-NUREG--25519) APPLICATION OF MULTIPLE TIMESTEP INTEGRATION METHOD IN SSC. Guppy, J.G. (Brookhaven National Lab., Upton, NY (USA)). 1979. Contract EY-76-C-02-0016. 5p. (CONF-790816--4). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The thermohydraulic transient simulation of an entire LMFBR system is, by its very nature, complex. Physically, the entire plant consists of many subsystems which are coupled by various processes and/or components. The characteristic integration timesteps for these processes/components can vary over a wide range. To improve computing efficiency, a multiple timestep scheme (MTS) approach has been used in the development of the Super System Code (SSC). In this paper: (1) the partitioning of the system and the timestep control are described, and (2) results are presented showing a savings in computer running time using the MTS of as much as five times the time required using a single timestep scheme.

- 321 (BNL-NUREG--25521) PHENOMENA AND SCENARIOS RELATED TO A LOSS-OF-HEAT-SINK ACCIDENT (WITH SCRAM) IN AN LMFBR. Bari, R.A.; Pratt, W.T.; Meyer, J.F. (Brookhaven National Lab., Upton, NY (USA); Nuclear Regulatory Commission, Washington, DC (USA)). 1978. Contract EY-76-C-02-0016. 10p. Dep. NTIS, PC A02/MF A01.

Several key phenomena and alternative scenarios which may follow a loss-of-heat-sink event, with scram, in an LMFBR are discussed. An event tree procedure is used to highlight the major branching points as they relate to controlling phenomena in the accident progression. For each of these branching points, a discussion of results of analyses carried out at BNL is presented. These discussions focus on the inherent safety capability features of the LMFBR concept as well as on particular design features of the Clinch River Breeder Reactor and of the Fast Flux Test Facility.

- 322 (BNL-NUREG--25522) CONTAINMENT RESPONSE TO POSTULATED EX-VESSEL CORE MELTDOWN ACCIDENTS IN THE FAST FLUX TEST FACILITY. Pratt, W.T.; Gasser, R.D.; Marchese, A.R. (Brookhaven National Lab., Upton, NY (USA); Nuclear Regulatory Commission, Washington, DC (USA). Div. of Project Management). 1979. Contract EY-76-C-02-0016. 12p. (CONF-790816--1).

Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

An evaluation of the containment response to ex-vessel core meltdown accidents in the Fast Flux Test Facility is discussed. Particular attention is given to the possibility of the molten fuel/steel mixture remaining in a molten form after entering the reactor cavity. For the molten pool configurations considered, significant H₂ concentrations were reached in the RCB before the reactor cavity floor was penetrated. The melt-front impacts the pressure transient, not through gas generation, but through penetration into the unlined subcavity resulting in an enhanced sodium-concrete reaction. Hence, if H₂ recombination is neglected, then steps would have to be taken to decrease the H₂ concentration before the RCB design pressure is reached.

- 323 (BNL-NUREG--25535) ONE-DIMENSIONAL MODEL FOR TRANSIENT BOILUP OF VOLUME-HEATED POOLS FOR LMFBR TRANSITION PHASE ANALYSIS. Ginsberg, T.; Chen, J.C. (Brookhaven National Lab., Upton, NY (USA); Lehigh Univ., Bethlehem, PA (USA). Dept. of Mechanical Engineering and Mechanics). 1979. Contract EY-76-C-02-0016. 14p. (CONF-790816--8). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

An analytical model is presented which describes the transient dispersion characteristics of fuel-steel boiling pool systems under decay heating conditions, with boundary heat losses. The spatial distribution of steel vapor (void fraction) and molten fluid is computed, together with the pressure-temperature histories of boiling pool systems. Local heat losses and sensible heating are included in the model for the vapor generation rate in the material balance equations. Model calculations are presented. Results indicate that while only a fraction of the decay heat source is useful for steel vaporization and dispersal, the potential nevertheless exists for dispersal of UO₂-steel pools to the limits of the available volume under decay heating conditions. Mild pressurization rates of 0.1 atm/s are computed for transition phase, closed-pool accident conditions.

- 324 (BNL-NUREG--25575) MICROWAVE HEATING SIMULATIONS OF FISSION ENERGY GENERATION IN VOLUME-BOILING POOL SYSTEMS. Makowitz, H.; Ginsberg, T. (Brookhaven National Lab., Upton, NY (USA)). 1979. Contract EY-76-C-02-0016. 8p. (CONF-790602--4). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

The transition phase of the (hypothetical) loss-of-flow accident in LMFBR's may be characterized by the temporary entrapment of boiling pools of molten fuel and steel within the original core boundaries. Experiments to investigate the multiphase flow and heat transfer characteristics of such systems have been performed and are planned in which simulant fluids are volume-heated by microwave electromagnetic radiation. It has been assumed that the microwave radiation provides a spatially uniform energy source per unit volume of liquid in multiphase flow geometries. It is known, however, that energy absorption by dielectrics in a microwave electromagnetic field is a function of geometry, dielectric properties, and wavelength of the radiation. At high power density, volume-heated boiling pools

exhibit a complex two-phase, liquid-vapor, geometric structure. Dispersed droplets, with diameters less than 1 mm, in a vapor continuum may coexist with continuous liquid structures of centimeter scale or greater. This study was performed to: (i) investigate the uniformity of microwave heating in boiling pool systems as a function of liquid geometry; and (ii) design an experimental system employing microwave heating in which the heat generation rate per unit volume of liquid is independent of geometric structure. The results of the study are presented.

- 325 (BNL-NUREG--25618) DYNAMIC SIMULATION OF LMFBR PLANT UNDER NATURAL CIRCULATION. Agrawal, A.K.; Guppy, J.G.; Madni, I.K.; Weaver, W.L. III. (Brookhaven National Lab., Upton, NY (USA)). 1979. Contract EY-76-C-02-0016. 42p. (CONF-790808--8). Dep. NTIS, PC A03/MF A01.

From 18. ASME national heat transfer conference; San Diego, CA, USA (Aug 1979).

An adequate dissipation of after-heat via natural circulation can be demonstrated by dynamic simulation of the entire LMFBR plant. Models needed to accomplish this task are briefly described in the paper. The resulting computer code, SSC-L, is then applied to simulate the response of the Clinch River Breeder Reactor Plant for a loss-of-electric power transient. Entire simulation is performed for 1800 seconds of the transient time. The decay heat removal via natural circulation is shown. The effect of internal flow redistribution in the reactor core is also demonstrated.

- 326 (BNL-NUREG--25714) COMPARISON OF MEASURED AND CALCULATED AVERAGE VOID FRACTION IN VOLUME-BOILING POOLS WITH INCLINED BOUNDARIES. Greene, G.A.; Jones, D.C. Jr.; Abuaf, N.; Schwarz, C.E. (Brookhaven National Lab., Upton, NY (USA)). 1979. Contract EY-76-C-02-0016. 9p. (CONF-790602--5). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

Accident analysis of an HCDA in an LMFBR often involves consideration and calculation of thermal hydraulic events related to boiling pools of fuel and steel mixtures in the disrupted core, as well as the subsequent motion of these two-phase mixtures out of the core. In many cases, the thermal-hydraulic state of the boiling fuel-steel or fuel pool may determine the initial or boundary conditions for subsequent calculations. Calculations of boundary heat losses are directly dependent on knowledge of the local and pool-averaged void fraction, and assessment of models to predict the average void fraction depends on the availability of these experimental data. Experiments have been performed to determine the distribution of boundary heat losses from volume-heated boiling pools and the pool-averaged void fraction.

- 327 (BNL-NUREG--26083) HEAT REMOVAL CHARACTERISTICS OF VOLUME-HEATED BOILING POOLS WITH INCLINED BOUNDARIES IN BUBBLY FLOW REGIME. Greene, G.A.; Jones, D.C. Jr.; Abuaf, N.; Schwarz, C.E. (Brookhaven National Lab., Upton, NY (USA)). 1979. Contract EY-76-C-02-0016. 10p. (CONF-790808--7). Dep. NTIS, PC A02/MF A01.

From 18. ASME national heat transfer conference; San Diego, CA, USA (Aug 1979).

The heat transfer characteristics of volume-heated boiling pools are of importance in the safety analysis of hypothetical core disruptive

accidents in liquid metal fast breeder reactors. Investigations into the heat transfer and hydrodynamic behavior of such volume-heated boiling pools have been few. The existing work in boiling pool heat transfer is reviewed, and current modeling efforts are described. The current experiment is discussed in detail, and significant improvements in experimental techniques are described. The data for local convective boundary heat transfer coefficient and pool-averaged void fraction are presented for the bubbly flow regime over a range of dimensionless power, $j/\text{sub } g^{00}/U_{00}$, up to unity, and wall angles of 90° (vertical), 75°, and 60°. The data are compared to existing correlations for local heat transfer and to a one-dimensional drift flux model for local and average void fraction. It is shown that the drift flux model predicted the experimental data for average void fraction in the bubbly flow regime well when based upon the net boiling power. However, it was demonstrated that the existing correlations for local and average boundary heat flux (based on Gustavson's data) underpredicted the present data by as much as a factor of 2 or more.

- 328 (BNL-NUREG--26131) IN-VESSEL NATURAL CIRCULATION DURING A HYPOTHETICAL LOSS-OF-HEAT-SINK ACCIDENT IN THE FAST FLUX TEST FACILITY. Perkins, K.R.; Bari, R.A.; Pratt, W.T. (Brookhaven National Lab., Upton, NY (USA)). May 1979. Contract EY-76-C-02-0016. 27p. (CONF-791205--13). Dep. NTIS, PC A02/MF A01.

From ASME winter annual meeting; New York, NY, USA (2 Dec 1979).

The capability to remove decay heat from the FFTF core via in-vessel natural circulation has been analyzed for the preboiling phase using a lumped parameter model. The results indicate that boiling will occur in the average fuel assembly for a wide spectrum of initial conditions which appear to be representative of the hypothetical loss-of-heat-sink accident. Two-phase pressure drop calculations indicate that, once the saturation temperature is reached, coolability can only be assured for decay heat levels which are less than 0.5% of the operating power. A review of the limited sodium boiling data indicates that boiling-induced natural circulation may support up to 4% of the operating power, but geometric atypicalities and a large degree of inlet subcooling for the existing data limit the applicability to the loss-of-heat-sink accident in FFTF.

- 329 (BNL-NUREG--26572) FEEDBACK CONTROL SYSTEMS FOR NON-LINEAR SIMULATION OF OPERATIONAL TRANSIENTS IN LMFBRs. Khatib-Rahbar, M.; Agrawal, A.K.; Srinivasan, C.S. (Brookhaven National Lab., Upton, NY (USA)). 1979. Contract EY-76-C-02-0016. 14p. (CONF-790816--48). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Feedback control systems for non-linear simulation of operational transients in LMFBRs are developed. The models include (1) the reactor power control and rod drive mechanism, (2) sodium flow control and pump drive system, (3) steam generator flow control and valve actuator dynamics, and (4) the supervisory control. These models have been incorporated into the SSC code using a flexible approach, in order to accommodate some design dependent variations. The impact of system nonlinearity on the control dynamics is shown to be significant for severe perturbations. Representative result for a 10 cent and 25 cent step insertion of reactivity and a 10% ramp

change in load in 40 seconds demonstrate the suitability of this model for study of operational transients without scram in LMFBRs.

- 330 (BNL-NUREG--50603) POST-ACCIDENT FUEL RELOCATION AND HEAT REMOVAL IN THE LMFBR. Kazimi, M.S.; Tsai, S.S.; Gasser, R.D. (Brookhaven National Lab., Upton, NY (USA)). Aug 1976. Contract EY-76-C-02-0016. 116p. Dep. NTIS, PC A06/MF A01.

Assessment of the dynamics of post-accident fuel relocation and heat removal is an important aspect of the evaluation of the consequences of a hypothetical accident in an LMFBR. Such an assessment is of particular importance in the evaluation of the post-accident radiological doses around the reactor site. In the present evaluation particular attention is given to the design features of the Clinch River Breeder Reactor Plant (CRBR). Fuel relocation and heat removal, assuming certain conditions have resulted in core disruption, are discussed. The discussion of events and phenomena involved in the relocation processes is centered around the resulting patterns of heat source distribution. The factors influencing fuel relocation and distribution in the inlet and outlet plena of the reactor vessel are discussed. The current technology of in-vessel heat removal is applied to the design of the CRBR reactor. Both fuel debris cooling limits and overall coolant flow in the reactor under natural convection conditions are explored. Some of the uncertainties in ex-vessel fuel behavior are addressed. In particular, the effect of melting the cavity bed on the rate of growth of a molten fuel pool is investigated.

- 331 (BNL-NUREG--50759) THERMO-FLUID MECHANICS OF VOLUME-HEATED BOILING POOLS. Greene, G.A.; Jones, O.C. Jr.; Schwarz, C.E. (Brookhaven National Lab., Upton, NY (USA)). Dec 1977. Contract EY-76-C-02-0016. 12p. Dep. NTIS (US Sales Only), PC A02/MF A01.

A program to study the behavior of volume-heated boiling pools has been undertaken at BNL in order to provide meaningful estimates of heat transfer mechanisms in such systems. Emphasis has been placed on developing appropriate scaling parameters to allow results to be applied to prototypical, hypothetical accident conditions in an LMFBR. It has been determined that, within experimental accuracy, a boiling pool seems to exhibit behavior much like a single-phase natural convection system with a much extended laminar film regime to local Rayleigh numbers approximately 10^{12} .

- 332 (BNL-NUREG--50773) ADVANCED THERMOHYDRAULIC SIMULATION CODE FOR TRANSIENTS IN LMFBRs (SSC-L CODE). Agrawal, A.K. (Brookhaven National Lab., Upton, NY (USA)). Feb 1978. Contract EY-76-C-02-0016. 331p. Dep. NTIS, PC A15/MF A02.

Physical models for various processes that are encountered in preaccident and transient simulation of thermohydraulic transients in the entire liquid metal fast breeder reactor (LMFBR) plant are described in this report. A computer code, SSC-L, was written as a part of the Super System Code (SSC) development project for the "loop"-type designs of LMFBRs. This code has the self-starting capability, i.e., preaccident or steady-state calculations are performed internally. These results then serve as the starting point for the transient simulation.

RESEARCH PROGRAMS. QUARTERLY PROGRESS REPORT, OCTOBER 1--DECEMBER 31, 1977. Romano, A.J. (comp.). (Brookhaven National Lab., Upton, NY (USA)). Jan 1978. Contract EY-76-C-02-0016. 214p. Dep. NTIS, PC A10/MF A01.

HTGR safety evaluation included studies on fission product release; materials, chemistry, and instrumentation; structural evaluation; and analytical safety evaluation. LMFBR safety evaluation included studies on accident sequences, technical coordination of structural integrity, and SSC code development and validation. LWR safety studies included thermal/hydraulic accident analysis, THOR code development, and stress corrosion cracking of PWR steam generator tubing.

- 334 (BNL-NUREG--50820) REACTOR SAFETY RESEARCH PROGRAMS. QUARTERLY PROGRESS REPORT, JANUARY 1--MARCH 31, 1978. Romano, A.J. (comp.). (Brookhaven National Lab., Upton, NY (USA)). Apr 1978. Contract EY-76-C-02-0016. 171p. Dep. NTIS, PC A09/MF A01.

Progress is summarized in the following areas: (1) gas reactor safety evaluation, (2) THOR code development, (3) foreign code review, (4) SSC code development, (5) LMFBR and LWR safety experiments, (6) fast reactor safety code validation, (7) stress corrosion cracking of PWR steam generator tubing, and (8) technical coordination of structural integrity.

- 335 (BNWL--510) DESIGN AND DEVELOPMENT QUALITY ASSURANCE REQUIREMENTS FOR THE FFTF. Albert, W.G. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 23 Oct 1968. Contract EY-76-C-06-1830. 55p. Dep. NTIS, PC A04/MF A01.

The document is presented to provide general management requirements for Pacific Northwest Laboratory (PNL) and contractor design and development quality assurance programs to assure the required quality level of the various items required for the FFTF. The document is applicable as imposed by the contract to FFTF contractors and subcontractors. The document is also applicable to PNL design and development activities related to the FFTF.

- 336 (BNWL--2216) INVESTIGATION OF COMBINED FREE AND FORCED CONVECTION IN A 2 X 6 ROD BUNDLE. Quigley, M.S.; McMonagle, C.L.; Bates, J.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1977. Contract EY-76-C-06-1830. 116p. Dep. NTIS, PC A06/MF A01.

The report documents a study designed to provide velocity and temperature data in a rod bundle geometry with low Reynolds number flows and severe radial power gradients. The data were then used to verify predictions by the COBRA computer program in the combined free and forced convection flow regime. Comparable flow conditions may occur in the radial blanket of a Liquid Metal Fast Breeder Reactor (LMFBR), in which radial power gradients up to 4 : 1 may exist across an assembly. Under low flow rate conditions the power gradient may cause significant buoyancy forces potentially resulting in flow redistribution within the radial assemblies. Consequently, there is a need to evaluate whether available thermal-hydraulic design codes can accurately predict the corresponding temperature and velocity profiles.

- 337 (BNWL-CC--2500) TECHNICAL BASES: FFTF FUEL/CLADDING DAMAGE CRITERIA. Hanson, J.E.; Ballard, E.O.; Bard, F.E. Jr.; Bullington, D.C.;

- 333 (BNL-NUREG--50785) REACTOR SAFETY

- Leggett, R.D. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Feb 1970. Contract EY-76-C-06-1830. 271p. Dep. NTIS, PC A50/MF A01.
- Specification of fuel damage criteria is important in evaluating and verifying the adequacy of safety margins for the FFTF. These margins are provided to meet the established safety objectives for the facility. The purpose of this report is to review the parameter limits specified as preliminary design data in the FFTF design safety criteria, and to identify the bases for each of these limit values. These bases provide a logical framework for discussion of the uncertainties involved as well as emphasize the more promising options for maximizing the return on investment of development funds.
- 338 (BR-E--180(Vol.1)) PLBR PHASE II: (FOLLOW-ON) REPORTS. (Burns and Roe, Inc., Oradell, NJ (USA)). 30 Sep 1977. Contract EY-76-C-03-1145. 408p. AT.
- Included are sections dealing with PLBR component seismic response characteristics and the response of the reactor cavity and primary heat transport system to the consequences of sodium pipe breaks. (DG)
- 339 (BR-E--180(Vol.2)) PLBR PHASE II: (FOLLOW-ON) REPORTS. VOLUME II. Byoun, T.; Celnik, J.; Babel, P.J. (Burns and Roe, Inc., Oradell, NJ (USA)). 30 Sep 1977. Contract EY-76-C-03-1145. 204p. AT.
- The following PLBR design projects are described: (1) radiation shielding analysis for the reactor cooling system, (2) primary heat transport system cell layout and model, (3) steam cycle design basis, and (4) reactor support ledge and pipe penetration cooling. (DG)
- 340 (BR-E--180(Vol.2)) PLBR PHASE II: (FOLLOW-ON) REPORTS. (Burns and Roe, Inc., Oradell, NJ (USA)). 30 Sep 1977. Contract EY-76-C-03-1145. 203p. AT.
- Engineering program objectives include work in the following areas: labyrinth shielding of primary piping; HTS cell layout; steam cycle design basis; reactor cavity structure gamma heating; and reactor support ledge and pipe penetration cooling.
- 341 (CE-FBR--77-26) EFFECT OF OPERATING TEMPERATURE ON LMFBR CORE PERFORMANCE. Noyes, R.C.; Bergeron, R.J.; di Lauro, G.F.; Kulwich, M.R.; Stuteville, D.W. (Combustion Engineering, Inc., Windsor, CT (USA)). 11 Apr 1977. Contract EY-76-C-02-2426-001. 165p. Dep. NTIS, PC A08/MF A01.
- The purpose of the study is to provide an engineering evaluation of high and low temperature LMFBR core designs. The study was conducted by C-E supported by HEDL expertise in the areas of materials behavior, fuel performance and fabrication/fuel cycle cost. The evaluation is based primarily on designs and analyses prepared by AI, GE and WARD during Phase I of the PLBR studies.
- 342 (CEA-BIB--226) CORROSION BY LIQUID METALS - APPLICATION TO LIQUID SODIUM. Lavielle, L. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides). Oct 1978. 72p. (In French). Dep. NTIS (US Sales Only), PC A05/MF A01.
- In this bibliographic review on the corrosion by liquid metals, the first part is

devoted to the theoretical aspects of the problem and the second part concerns the corrosion of steels by liquid sodium, as example. Obvious the numerous works now published, the mechanisms are still leaving bad known.

- 343 (CEA-CONF--3749) SEISMIC ANALYSIS FOR THE SUPER-PHENIX REACTOR. Lefevre, J.C.; Livolant, M.; Lucenet, G. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. des Etudes Mecaniques et Thermiques). 1976. 11p. (CONF-761006--46). Dep. NTIS (US Sales Only), PC A02/MF A01.
- From Nuclear science, scintillation and semiconductor counter symposium; New Orleans, LA, USA (20 Oct 1976).
- The seismic analysis performed for SUPER-PHENIX is presented. The analysis consists essentially of: selecting reference ground motions based on the seismic risks at the site; establishing general earthquake safety criteria; studying the response of buildings, systems and components, using the most appropriate methods; establishing design and acceptance criteria in order to verify the validity of the conception and the reliability of the equipment.
- 344 (CEA-CONF--3834(Pt.2)) TRIPOLI 01, A THREE-DIMENSIONAL POLYKINETIC MONTE CARLO PROGRAM. PT.2. CONSTANT DATA INPUT AND RESULTS OBTAINED FOR A COMPLEX BENCH-MARK EXAMPLE. Baur, A.; Bourdet, L.; Gonnord, J.; Nimal, J.C.; Vergnaud, T. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Eau). Jan 1977. 165p. (In French). (CONF-7610162--P2). Dep. NTIS (US Sales Only), PC A08/MF A01.
- From Seminar workshop on the Monte Carlo code TRIPOLI; Las Vegas, NV, USA (17 Oct 1976).
- Some properties of the sub-routines included in TRIPOLI for the constant data input are briefly reviewed. The bench-mark example then proposed is a problem of the contour lines of a neutron-opaque shield. The example is derived from activation calculations for the secondary sodium in a fast reactor provided with an integrated shield of Phenix type. The dimensions have been reduced so as the geometry to be simplified, and the constant data load reduced. Besides the rate of ^{24}Na formation by capture in the exchanger region, the attenuation of various detector responses all along the neutron trajectory was taken into account. The detectors chiefly used were: ^{103}Rh activation from (n,n') reactions for fast neutron detection, the activation Mn detectors, either bare or under cadmium, for detecting thermal and epithermal neutrons respectively.
- 345 (CEA-CONF--3848) KEY SAFETY ISSUES FOR FAST BREEDER REACTORS. Tanguy, P.; Petit, J.; Justin, F. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. de Surete Nucleaire). 1976. 8p. (CONF-761001--). Dep. NTIS (US Sales Only), PC A02/MF A01.
- From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).
- Safety evaluations made on the commercial 1200 Mwe Creys-Malville LMFBR have shown a number of key safety issues: fuel cladding integrity, core stability, core coolability, core monitoring, shutdown systems, core disruptive accident and containment. Not one of them seems unsolvable, but studies and tests appear still necessary, especially if it is thought in simplifying the system of increasing

the unit size of the plant.

- 346 (CEA-CONF--3849) SAFETY TEST FACILITIES. NEEDS AND CONCEPTS. A FRENCH EVALUATION. Tretiakoff, O.; Bailly, J. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. de Surete Nucleaire). 1976. 7p. (CONF-761001--). Dep. NTIS (US Sales Only), PC A02/MF A51.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The fuel behaviour of LMFBRs in the event of an accident has been tested in-pile in the SCARABEE program (local blockage, sudden flow reduction and pump coast-down at constant power). These tests will be carried on in the framework of an international cooperation on irradiated fuels: this is the purpose of the CABRI and SCARABEE N programs. All those studies should enable to assess safety margins between accident conditions and the technical specifications of the reactor. The paper explains how a logical set of simple observations has led to the present state of the Cadarache in-pile experimental safety program and how it may help to find the way in a dense forest of both technical and psychological difficulties.

- 347 (CEA-CONF--3890) NUCLEAR SAFETY RESEARCH IN FRANCE. Tanguy, P. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. de Surete Nucleaire). 1976. 22p. (CONF-761254--1). Dep. NTIS (US Sales Only), PC A02/MF A01.

From Seminar on nuclear safety; Tokyo, Japan (17 Dec 1976).

As a consequence of the decision of choosing light water reactors (PWR) for the French nuclear plants of the next ten years, a large safety program has been launched referring to three physical barriers against fission product release: the fuel element cladding, main primary system boundary and the containment. The parallel development of French-designed fast breeder reactors involved safety studies on: sodium boiling, accidental fuel behavior, molten fuel-sodium interaction, core accident and protection, and external containment. The rapid development of nuclear energy resulted in a corresponding development of safety studies relating to nuclear fuel facilities. French regulations also required a special program to be developed for the realistic evaluation of the consequences of external aggressions, the French cooperation to multinational safety research being also intensive.

- 348 (CEA-CONF--3987) METALLURGICAL ASPECTS OF THE WELDABILITY OF CR-MO STABILIZED FERRITIC STEELS. Colombe, G.; Petrequin, P.; Debriez, S.; Donati, J.R.; Zacharie, C. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. de Physique Nucleaire). 1977. 16p. (CONF-770545--4). Dep. NTIS (US Sales Only), PC A02/MF A01.

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

Results of tests performed to compare the tendency of 2.25 Cr-Mo and 9 Cr-2 Mo steels stabilised with niobium and niobium + vanadium, to different cracking phenomena susceptible to occur during welding, stress-relieving and service in the conditions of fast breeder reactors are presented.

- 349 (CEA-CONF--3994) HOW LOCAL SODIUM BOILING IN ROD BUNDLES UNDER REALLY STEADY

STATE BOUNDARY CONDITIONS IS ACHIEVED ON THE CFNA LOOP. Menant, B. (CEA Centre d'Etudes Nucleaires de Grenoble, 38 (France). Dept. de Transfert et Conversion d'Energie). 1977. 10p. (CONF-770640--6). Dep. NTIS (US Sales Only), PC A02/MF A01.

From 7. liquid metal boiling working group meeting; Petten, Netherlands (1 Jun 1977).

A good knowledge on sodium boiling under steady state boundary conditions is a necessary basis for the analysis of sodium boiling in the area of the fast reactor safety. The laps of time necessary to establish permanent local boiling in rod bundles is sometimes longer than a few seconds. Therefore steady state boundary conditions (power, flow rate, inlet temperature, outlet pressure) can be maintained as long as desired on the CFNA loop. Several cycles of measurements are performed during each run in order to make sure that a permanent thermohydraulic regime is reached everywhere in the test section.

- 350 (CEA-CONF--3995) NUCLEATION, DRY-OUT AND BOILING BEHIND A SODIUM-TIGHT LOCAL BLOCKAGE IN A 9-PIN BUNDLE WITH HELICAL WIRE SPACERS. Menant, B. (CEA Centre d'Etudes Nucleaires de Grenoble, 38 (France). Dept. de Transfert et Conversion d'Energie). 1977. 15p. (CONF-770640--4). Dep. NTIS (US Sales Only), PC A02/MF A01.

From 7. liquid metal boiling working group meeting; Petten, Netherlands (1 Jun 1977).

Permanent sodium boiling tests were performed with a 19 wire-wrapped pin bundle in which the 12 central subchannels were sodium-tightly blocked. No risk of boiling was evidenced with nominal cooling and power, but the boiling which occurred when the flow rate was sufficiently restricted was very damagable. Indeed dry-out occurred before any evidence of vapor bubbles in the flow could be obtained with classical means of investigation (i.e. electromagnetic flowmeters and pressure transducers). The interpretation, leading to a model of nucleation and dry-out around hot spots in the wake of the blockage, is corroborated by the analysis of thermal and acoustical noise.

- 351 (CEA-CONF--3996) SODIUM VOIDING IN AN ANNULAR CHANNEL HEATED BY A SINGLE PIN IN OVERPOWER ACCIDENT SIMULATIONS. Seiler, J.M. (CEA Centre d'Etudes Nucleaires de Grenoble, 38 (France). Dept. de Transfert et Conversion d'Energie). 1977. 8p. (CONF-770640--5). Dep. NTIS (US Sales Only), PC A02/MF A01.

From 7. liquid metal boiling working group meeting; Petten, Netherlands (1 Jun 1977).

In a simulation of overpower accidents affecting a fast breeder core, experiments on sodium boiling were performed in a single vertical channel of annular cross section (10.65 mm I.D.) indirectly heated by a high flux heater pin (8.65 mm O.D.). Measurements were made with a high speed acquisition system. The experimental conditions were: Pressure of cover gas: 1 kg/cm² abs, Mass Flow rate of sodium: 25 g/s, Inlet temperature: 400°C, Steady state outlet temperature: 800°C, Transient power supply: 90 KW up to 250 KW. For the first expulsion, the results revealed two important points: variable initial superheat with corresponding pressure pulse, and after superheat resorption, thermal conditions prevail and the boiling process followed equilibrium conditions. After heating cut off, rewetting pressure pulses were observed.

- 352 (CEA-CONF--4013) EFFECT OF STRUCTURE AND ALLOYING ELEMENTS ON VOID FORMATION IN

AUSTENITIC STEELS AND NICKEL ALLOYS. Levy, V.; Azam, N.; Le Naour, L.; Didout, G.; Delaplace, J. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. de Technologie). 1977. 19p. (CONF-770641--). Dep. NTIS (US Sales Only), PC A02/MF A01.

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

In the development of the fast breeder reactors the phenomenon of metal swelling due to the formation of radiation induced voids is a large problem. In the complex alloys small fluctuations in composition can have a considerable effect on swelling and a great deal of investigation on the effect of both major and minor alloying elements is needed to be able to predict swelling. To provide more insight a research program involving irradiation of both commercial or specially cast alloys by 500 keV Ni⁺ ions or 1 MeV electrons has been developed. The major results are presented.

353 (CEA-CONF--4014) FUNDAMENTAL ASPECTS OF THE EVOLUTION OF, AND PHASE CHANGES IN, METALS AND ALLOYS UNDER IRRADIATION. Martin, Georges; Bocquet, J.-L.; Barbu, Alain; Adda, Yves. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. de Technologie). 6 Jun 1977. 16p. Dep. NTIS (US Sales Only), PC A02/MF A01.

From International conference on radiation effects in breeder reactor structural materials; Scottsdale, AR, USA (19 Jun 1977).

The quantitative prediction and the simulation of irradiation damage present a considerable interest for the technology of fast neutron and fusion reactors. However, at present time, even qualitative predictions are not always possible when the dose rate varies over a relatively large range; for example the formation of vacancy loops takes place instead of that of voids, precipitation occurs in under-saturated solid solutions etc... Therefore, in addition to a microscopic description of the phenomena, it seems important to elaborate an irradiation damage phenomenology. The results of a systematic experimental study of dose rate effects for radiation induced precipitation in Ni Si solid solution alloys are reported. The existence of dose rate thresholds for this phenomenon is clearly established. The composition dependence of this thresholds permits the definition of a solubility limit under irradiation which is both temperature and dose rate dependent. Finally, the various theoretical approaches to radiation induced precipitation are discussed.

354 (CEA-CONF--4192) UTILIZATION OF THE EXPERIMENTAL REACTOR OSIRIS FOR THE STUDY AND THE DEVELOPMENT OF FUELS OF THE FAST NEUTRON REACTOR TYPE. Marcon, M.; Faugere, J.L.; Genthon, J.P.; Maillot, R. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Services des Piles). 1977. 21p. (In French). (CONF-771189--5). Dep. NTIS (US Sales Only), PC A03/MF A01.

From Symposium on the future of research reactors; Grenoble, France (Nov 1977).

Nuclear fuel tests for the fast neutron reactor type have been carried out at the Osiris reactor: thermal study of (U,Pu)₂O₃ oxide by measurement with thermocouples in the core of the fuel pellet; study of the effects of power cycling on nuclear fuel; study of the mechanical interactions between oxide and cladding by measurement of the cladding deformation during irradiation.

355 (CEA-CONF--4212) BREEDER REACTORS. Gollion, H.; Lucenet. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides). 1977. 52p. (In French). (CONF-7703127--1). Dep. NTIS (US Sales Only), PC A04/MF A01.

From One-day information meeting on nuclear power plants; Grenoble, France (9 Mar 1977).

The reasons for the development of fast reactors are briefly reviewed (a propitious neutron balance oriented towards a maximum uranium burnup) and its special requirements (cooling, fissile material density and reprocessing) discussed. The three stages in the French program of fast reactor development are outlined with Rapsodie at Cadarache, Phenix at Marcoule, and Super Phenix at Creys-Malville. The more specific features of the program of research and development are emphasized: kinetics and the core, the fuel and the components.

356 (CEA-CONF--4339) NUMERICAL METHOD FOR A TRANSIENT TWO-FLUID MODEL. Le Coq, G.; Libmann, M. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Eau). 1978. 15p. (CONF-7805170--2). Dep. NTIS (US Sales Only), PC A02/MF A01.

From 2. Phase Flow Group meeting; Stockholm, Sweden (28 May 1978).

The transient boiling two-phase flow is studied. In nuclear reactors, the driving conditions for the transient boiling are a pump power decay or/and an increase in heating power. The physical model adopted for the two-phase flow is the two fluid model with the assumption that the vapor remains at saturation. The numerical method for solving the thermohydraulics problems is a shooting method, this method is highly implicit. A particular problem exists at the boiling and condensation front. A computer code using this numerical method allow the calculation of a transient boiling initiated by a steady state for a PWR or for a LMFBF.

357 (CEA-CONF--4384) CALCULATION OF STRUCTURES BY A METHOD OF SUBSTRUCTURES DEFINED BY THEIR EIGENMODES. LINEAR AND NONLINEAR APPLICATIONS. TRISTANA CODE. Jeanpierre, F.; Livolant, M. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. des Etudes Mecaniques et Thermiques). 1978. 26p. (In French). (CONF-7806167--3). Dep. NTIS (US Sales Only), PC A03/MF A01.

From Summer meeting: shocks and vibrations; Lyon, France (13 Jun 1978).

An efficient method for complex structures dynamical analysis is the method of independent substructures connected by mechanical links, the movement of the substructures being defined on their natural modes basis. This method allows the calculation of non linear links like shocks or fretting. Studied in CEA/DEMT it has been introduced in the TRISTANA code. Some applications are presented in this report: tests of the method or application to nuclear reactor internals vibrational calculations.

358 (CEA-CR--13, pp vp) EFFECTS OF LOAD FOLLOWING ON THE CORE AND THE COMPONENTS OF THE SUPER PHENIX SODIUM CIRCUITS. Acket (Compagnie d'Ingenierie pour les Reacteurs au Sodium (CIRNA), Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)); Decuyper; Reynaud (Societe Centrale Nucleaire Europeenne a Neutrons Rapides (NERSA), 69 - Lyon (France)); Quinton (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance

(France). Dept. des Reacteurs a Neutrons Rapides). [nd]. (In French).

Nuclear power plant control problems associated with loads following and network transients.

After describing the installation, a description is given of the objectives for the load following of the Creys-Malville power station, as well as of the execution, study means and tests.

- 359 (CEA-N--2075) EXPERIMENTAL TESTS ON BUCKLING OF ELLIPSOIDAL VESSEL HEADS UNDER INTERNAL PRESSURE. Alix, M.; Roche, R. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. des Etudes Mecaniques et Thermiques). Jan 1979. 117p. (In French). Dep. NTIS (US Sales Only), PC A06/MF A01.

Seventeen heads made out of metal sheets -by cold working- were tested. Three different metals were used - carbon steel, austenitic steel, and aluminium alloy. Nominal dimensions were: diameter D 500 mm height H 50 and 100 mm thickness to diameter ratio t/D in the range 0.001-0.005. The heads had a good axisymmetric shape, but that the thickness was varying along the ellipse. Material characteristic of each head was given by a tensile test (strain-stress curve). The obtained results are mainly the pressure deflexion recordings, strain measurements and visual observations of the geometrical changes. For thin heads, buckling is a very fast event and the first folding occurs suddenly, with a strong perturbation on the pressure-deflexion curve. For the thickest heads, circular waves are slowly forming. In all of these tests, yielding occurred before buckling and it was possible to increase the pressure beyond the first buckling pressure without failure. The experimental results agree very well ($\pm 5\%$ except one head) with the empirical formula $P_{sub}(c) = 70000 \cdot (\sigma_y + \sigma_u / 2) \cdot (t/D)^{5/3} \cdot ((D/H)^2 - 8)^{2/3}$. The following notations being used: $P_{sub}(c)$: critical buckling pressure; σ_y : yield strength; σ_u : ultimate stress (same unit); t: knuckle thickness; D: mean diameter; H: height (same unit).

- 360 (CEA-R--4881) MANUFACTURE OF THE SECOND CORE OF RAPSODIE - TYPE 2 A - FORTISSIMO VERSION. Mandicourt, L. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. de Developpement des Elements Combustibles). Nov 1977. 88p. (In French). Dep. NTIS (US Sales Only), PC A05/MF A01.

This report concerns the fuel manufacturing of the second core of RAPSODIE reactor 2 A type, FORTISSIMO version. This fuel is a mixed oxide of Plutonium and highly enriched Uranium. The manufacturing facility which is used, started in 1969, is essentially designed for the production of fuels with high percentage of fissile material. It includes devices used to produce the first RAPSODIE fuel and new machines resulting from technological improvements obtained during previous manufactures. The manufacturing process is described after a recall fuel physical characteristics. A final quantitative balance for the whole fabrication is presented. The results obtained on fuel pellets and pins are discussed with starting from the fabrication process and the changes which may have occurred or arose from the use of different devices. A more theoretical approach of some particular problems is presented: the multiphasage, the amount of oxygen and the porosities. A general conclusion is given on the main theoretical and technical problems met on the manufacture of

nuclear fuel of this type.

- 361 (CEA-R--4905) STUDY OF THE Na-C-O AND Na-H-O TERNARY SYSTEMS IN THE SODIUM RICH CORNER. Maupre, J.-P. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides; Aix-Marseille-1 Univ., 13 - Marseille (France)). Feb 1978. 367p. (In French). Dep. NTIS (US Sales Only), PC A15/MF A01.

Thesis.

The purpose of this study is to provide a contribution to the understanding of the sodium - carbon - oxygen and sodium - hydrogen - oxygen ternary systems in the sodium rich corner. In order to do this the Na-NaH-Na₂O-NaOH phase diagram was completed and the Na-Na₂O-Na₂CO₃-C phase diagram was outlined. This work is made up of two parts. The first is devoted to a critical literature survey essential to establish correct phase diagrams. The second is an experimental study followed by a discussion collating our finding to the literary data. The basic experimental technique used is differential thermal analysis (DTA) but it has been completed by quenching, X-ray and chemical analysis methods. The proposed phase diagrams imply that Na-NaH-Na₂O-NaOH and Na-Na₂O-Na₂CO₃-C systems are reciprocal ternary systems. Temperatures of stable pairs reversal are respectively 410 and 690°C. The stable pairs are Na-NaOH and Na-Na₂CO₃ at elevated temperature, Na₂O-NaH and Na₂O-C at low temperature.

- 362 (CEA-R--4911) FAST NEUTRON REACTOR CORE RESEARCH AT THE C.E.A. Chaudat, J.P. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides). May 1978. 121p. (In French). Dep. NTIS (US Sales Only), PC A06/MF A01.

This report covers all physical studies of fast neutron reactors carried out by the C.E.A., to provide basic data (multi-group cross sections) and computer methods which may be used to calculate nuclear power plant neutron properties with the precision required by the project. The approach adopted to establish the basic data used in all core calculations is described in greater detail: choice of a reference procedure for basic mode calculations (CARNAVAL set), choice of particular experimental programs to reduce uncertainties in connection with the formula set, adjustment of cross sections on integral parameters measured on critical experiments. The development of the formula set is closely connected with the project requirements; hence the set is modified with respect to the core characteristics of the power plant studied. Following an explanation of how the CARNAVAL III and IV formula sets -used for PHENIX and SUPER-PHENIX respectively- were derived, current studies for heterogeneous cores are described.

- 363 (CEGB-RD/B/N--3414) FRODA - A PROGRAM FOR THE ANALYSIS OF FREQUENCY DYNAMICS EXPERIMENTS. Emmett, J.C.A. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). May 1976. 40p. Dep. NTIS (US Sales Only), PC A03/MF A01.

The FRODA program has been produced for the analysis of dynamics experiments in the frequency domain. Its two main functions are the processing of experimental frequency responses and the fitting of theoretical frequency responses from various simple mathematical models to experimental plant. Two

types of model can be used: (i) empirical models of which there are three different formulations of the standard polo-zero type of model and (ii) physics models which consist of a set of linearized algebraic and first order differential equations. The FRODA calculation is divided into separate tasks which may be performed singly or in any feasible combination. These tasks have all been used successfully in a simulated analysis of the P.F.R. Oscillator Experiment and it is intended to use FRODA to analyse the real experimental data when it becomes available. FRODA has been written in a general form for use for evaluation and fitting of a wide variety of simple plant models in the frequency domain. Other nuclear applications have been the analysis of noise measurements on a Magnox reactor and the setting up of a simple AGR core model for simulating a dynamic experiment to measure reactivity coefficients. A general description of FRODA and instructions on how to use the program is given.

- 364 (CEGB-RD/B/N--3666) RECENT ADVANCES IN UNDERSTANDING LARGE SCALE VAPOUR EXPLOSIONS. Board, S.J.; Hall, R.W. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Apr 1977. 32p. (CONF-760328--7). Dep. NTIS (US Sales Only), PC A03/MF A01.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

Recent advances in the description of large scale efficiency vapour explosions indicate that three stages must be considered: setting up a quasi-stable initial configuration; triggering of this configuration; and propagation of the explosion. Each of these stages is considered in turn, the relevant experimental information and theory being reviewed to see to what the extent the requirement for energetic explosions and the physical processes satisfying the requirements are understood. Recent models for large scale explosions have helped to achieve an improved understanding, particularly of the propagation phase. Particular attention is paid to a suggested criterion for explosivity that the contact temperature should exceed the temperature of spontaneous nucleation of the coolant, for this criterion sodium and UO₂ in particular, are not explosive.

- 365 (CEGB-RD/B/N--3810) LOCAL BLOCKAGE STUDIES IN A HEATED MULTIPHASE WATER EXPERIMENT WITH FAST REACTOR GEOMETRY. Clare, A.J. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). May 1977. 40p. Dep. NTIS (US Sales Only), PC A03/MF A01.

An important stage in studying the effects of local blockages in LMFBR subassemblies concerns the prediction of the elevated coolant temperatures in the downstream recirculating wake, and in particular the size of blockage which would cause boiling of the coolant. The development of both sophisticated models of the flow and heat transfer in the wake (such as the UK code SABRE), and simpler empirical methods, requires adequate experimental support including measurements of wake temperature patterns in large cluster experiments. Results are reported of a detailed study of wake temperature distributions produced in single-phase flow by blockages in a 283 pin (91 heated) water experiment. Two central blockages (7.2% and 12.8% of total flow area blocked) and two corner blockages (5.4% and 8.7% blocked) have been studied.

- 366 (CEGB-RD/B/N--3982) MEASUREMENTS OF TEMPERATURE FLUCTUATIONS IN THE MIXING OF HOT AND COLD AIR JETS. Sumner, V.W. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Mar 1977. 41p. Dep. NTIS (US Sales Only), PC A03/MF A01.

In order to assess the effect of the mixing of 'hot' and 'cold' jets of sodium on structures in the above-core region of the fast reactor, temperature fluctuations have been measured in an experiment consisting of a heated jet of air surrounded by six unheated jets. Temperature spectra obtained from the experiment showed no strong peaks or bands. In considering the effect of thermal cycling of the above-core structures, it is the higher strain values at low frequencies which will be more limiting than the smaller values at high frequencies, due to the nature of strain-lifetime curves. Thus the spectra have been summarised using a low-frequency level and a cut-off frequency at which this level has fallen by an order of magnitude. Attenuation of temperature fluctuations due to the high thermal conductivity of sodium or by boundary layer effects has been considered; however, in the low-frequency, high-energy region of the spectra, little attenuation can be expected.

- 367 (CEGB-RD/B/N--4029) TECHNIQUE FOR MEASURING HYDROGEN AND WATER IN INERT GASES AND THE HYDROGEN CONCENTRATION IN LIQUID SODIUM. Smith, C.A. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Apr 1978. 26p. Dep. NTIS (US Sales Only), PC A03/MF A01.

A method is described of measuring the hydrogen and water content of an inert gas. It is based upon the use of an electrochemical oxygen cell and has a high sensitivity at low hydrogen and water levels. The following possible applications of the method are described together with supporting experimental measurements: improving the sensitivity and range of the present PFR secondary circuit hydrogen detection instruments; the measurement of hydrogen diffusion coefficients in steels; the measurement of waterside corrosion rates of boiler steels; on-line monitoring of waterside boiler corrosion. Attention is given to the characteristics of diffusion barriers in relation to the first and last of these.

- 368 (CEGB-RD/B/N--4194) FISSION GAS BUBBLE DISTRIBUTION IN A MIXED OXIDE FAST REACTOR FUEL PIN. Baker, C. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Dec 1977. 11p. Dep. NTIS (US Sales Only), PC A02/MF A01.

The fission gas bubble distribution has been studied in a mixed oxide fast reactor fuel pin irradiated in DIDO MTR to 2.8% burn-up at centre and surface temperatures of 2000 and 1000°C. The intragranular fission gas bubbles are very small (< 6 mm diameter) and this is a consequence of the high re-solution rate at fast reactor ratings. The bubbles nucleate heterogeneously and linear arrays of bubbles, due to nucleation on fission tracks, are observed up to irradiation temperatures of 1900°C. At 1980°C approximately equal to 4% of the fission gas produced is present in intragranular bubbles. There is no definite evidence for gas bubble mobility or coalescence. Apart from any effects of columnar grain growth, fission gas release in fast reactor fuel pins seems to occur predominantly by the diffusion of single gas atoms, at least up to irradiation temperatures of 2000°C.

- 369 (CEGB/RD/B/N--4219) CREEP IN SODIUM: A REVIEW. Charnock, W.; Cordwell, J.E. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Mar 1978. 33p. (FRDC/MWP/P--(78)752; FRDC/MWP/MPSG/P--(78)172). Dep. NTIS (US Sales Only), PC A03/MF A01.

The available information on the creep of austenitic, ferritic, and Alloy-800 type steels in liquid sodium is critically reviewed. The creep properties of stainless steels can be affected by element transfer and corrosion. At reactor structural component temperatures environmental effects are likely to be less important than changes due to thermal ageing. At high clad temperatures (700°C) decarburization may cause the loss of strength and ductility in unstabilized steels while cavity formation may cause embrittlement in stabilized steels. In some experiments Alloy 800 properties are found to deteriorate while in others they are enhanced. Low alloy ferritic steels tend to decarburize in sodium at temperatures greater than approx. 500°C and this leads to loss of strength and an increase in ductility. High alloy ferritics are immune to this effect and appear to be able to tolerate a degree of carburization. Although in general the incidence of intergranular cracking may be enhanced in liquid sodium the mechanical consequences are not significant and the evidence for the existence of an embrittlement effect not associated with element transfer or corrosion is weak. Stress and strain may enhance element transfer at crack tips. However, in real cracks the gettering or supply action of the crack faces conditions the chemistry of the crack sodium and protects the crack tip from element transfer. Thus creep crack extension rates should be independent of changes in bulk coolant chemistry.

- 370 (CEGB/RD/B/N--4241) TERTIARY CREEP OF SOLUTION TREATED AISI 316 STAINLESS STEEL IN AIR AND IN SODIUM AT 550°C. Cordwell, J.E. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Feb 1978. 21p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Reported observations of tertiary creep embrittlement in short term tests on AISI 304 stainless steel at 550°C in sodium led to the present series of short duration creep tests on AISI 316 stainless steel at the same temperature. Neither the duration of tertiary creep nor the strain associated with it appeared to be systematically affected by the environment. However, both these quantities were small making any embrittling effect difficult to detect. Differences in the strains produced on initial loading may have been responsible for increased creep life in the sodium exposed specimens.

- 371 (CEGB/RD/B/N--4334) REMOVAL OF RADIOACTIVE PRODUCTS FROM LIQUID METAL COOLANTS OF LMFBRs UNDER NORMAL OPERATION. Hooper, A.J. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Aug 1978. 22p. Dep. NTIS (US Sales Only), PC A02/MF A01.

The levels of activity which will accumulate on components in the primary circuit of a liquid metal cooled fast breeder reactor as a result of deposition of activated corrosion products, fission products, and fuel particles from the liquid metal coolant must be minimized to facilitate maintenance and decontamination procedures. The information available in the open literature up to the end of 1977 relating to removal of radioactive products from the liquid metal coolants of LMFBRs and to the

mechanisms of deposition of these species is reviewed. Particular emphasis is placed on information emanating from operating LMFBRs.

- 372 (CEGB/RD/B/N--4338) EFFECT OF VARIATIONS IN CARBON ACTIVITY ON THE CARBURIZATION OF AUSTENITIC STEELS IN SODIUM. EXPLORATORY TESTS. Gwyther, J.R.; Hobdell, M.R.; Hooper, A.J. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Jul 1978. 23p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Experience has shown that the liquid sodium coolant of fast breeder reactors is an effective carbon-transport medium; the resulting carburization of thin austenitic stainless steel components (eg IHX and fuel cladding) could adversely affect their mechanical integrity. The degree and nature of steel carburization depend, inter alia, on the carbon activity of the sodium environment. Exploratory tests are described in which specimens of austenitic stainless steel were carburized in sodium, the carbon activity of which was continuously monitored by a BNL electrochemical carbon meter. The sodium carbon activity was initially high, but decreased with time, simulating conditions equivalent to plant start-up or coolant clean-up following accidental oil ingress. The extent and nature of steel carburization was identified by metallography, electron microscopy, X-ray crystallography and chemical analysis.

- 373 (CENC--1397) POOL TYPE IHX STUDY FOR THE LARGE COMPONENT DEVELOPMENT PROGRAM. Sowers, M.G.; DeFur, D.D. (Combustion Engineering, Inc., Chattanooga, TN (USA). Nuclear Components Dept.). 30 Sep 1979. Contract W-31-109-ENG-38. 129p. AT.

Pool type IHX designs for both 430 MWt and 645 MWt ratings should be feasible and practical with careful engineering studies of potential problems. C-E drawings show preferred configurations. The principle areas of concern are uncertainties associated with design latitude, flow distribution and seismic response. These and other less significant concerns are functions of IHX size; therefore, there is a higher degree of confidence for the smaller units. It is recommended that development programs be carried out to reduce the design uncertainties.

- 374 (CONF-681093-- pp 8.1-8.9) SEFOR: THE MOST PROMISING CONCEPT. Kerns, R.E. (Oklahoma Gas and Electric Co., Oklahoma City). 1968.

From Frontiers of power technology conference; Stillwater, OK, USA (28 Oct 1968). A brief survey is given of the planning and construction of the SEFOR reactor. Fuel element design and core configuration are examined. (DG)

- 375 (CONF-681093-- pp 9.1-9.51) UNIQUE FEATURES OF A SODIUM-COOLED FAST-BREEDER REACTOR. Ash, E.B. (North American Rockwell Corp., Canoga Park, CA). 1968.

From Frontiers of power technology conference; Stillwater, OK, USA (28 Oct 1968). The paper describes the characteristics of the liquid-metal cooled fast-breeder reactor (LMFBR). It is rapidly emerging as one of the major future energy sources throughout the world. The paper discusses the design features of this concept and describes the unique features which distinguish it from the more familiar light-water reactor (LWR). It highlights these features from a utility

viewpoint and emphasizes some significant advantages.

- 376 (CONF-760673-- pp 253-289) NEUTRON SWELLING OBSERVATIONS ON AUSTENITIC STAINLESS STEELS IRRADIATED IN EBR-II. Flinn, J.E. (Argonne National Lab., Idaho Falls, IA); Kenfield, T.A. 1976.

From Workshop on correlation of neutron and charged particle damage; Oak Ridge, TN, USA (8 Jun 1976).

Recent neutron swelling results on annealed types 304L, 304, and 316 stainless steels are presented. The results consist of percent volume changes as a function of irradiation temperature and fast fluence. The respective swelling data bases for each steel are not sufficient to provide a quantitative comparison between the three stainless steels. The most pronounced difference observed is their peak swelling temperatures, where 304 exhibits the lowest temperature (approximately 450°C) and 316 the highest (approximately 600°C). Influence due to cold-worked conditions on 304 and 316 stainless steels are examined. The 30% cold-worked condition for the two alloys showed a definitive decrease in swelling when compared to the respective annealed materials. Some preliminary results on enhanced swelling (possible stress effects) and irradiation history or memory effects on 304L stainless steel driver fuel cladding are also given.

- 377 (CONF-760673-- pp 291-311) APPLICATIONS OF SIMULATION EXPERIMENTS IN LMFBR CORE MATERIALS TECHNOLOGY. Appleby, W.K. (General Electric Co., Sunnyvale, CA). 1976.

From Workshop on correlation of neutron and charged particle damage; Oak Ridge, TN, USA (8 Jun 1976).

The development of charged particle bombardment experiments to simulate neutron irradiation induced swelling in austenitic alloys is briefly described. The applications of these techniques in LMFBR core materials technology are discussed. It is shown that use of the techniques to study the behavior of cold-worked Type-316 was instrumental in demonstrating at an early date the need for advanced materials. The simulation techniques then were used to identify alloying elements which can markedly decrease swelling and thus a focused reactor irradiation program is now in place to allow the future use of a lower swelling alloy for LMFBR core components.

- 378 (CONF-760735--1) LEAK DETECTION SYSTEM DESIGN AND OPERATING CONSIDERATIONS FOR THE US-CRBRP. Kruger, G.B.; Eng, K.Y.; Kelly, W.L. (General Electric Co., Sunnyvale, CA (USA); Energy Research and Development Administration, Washington, DC (USA)). 1976. Contract EY-76-C-15-2395. 23p. Dep. NTIS, PC A02/MF A01.

From US/USSR steam generator seminar; Dimitrovgrad, USSR (26 Jul 1976).

Diffusion membrane type hydrogen detectors are provided for monitoring the sodium exiting each evaporator and superheater in the Clinch River Breeder Reactor Plant. These detectors allow detection of small water to sodium leaks and provide the plant operator with an early warning signal. Hydrogen detectors are located at the exit sodium streams of each steam generator module, the vent from the module semi-stagnant region, the cold leg piping, and in an intermediate system sodium expansion tank cover gas region. In addition, an electrochemical oxygen detector is located in the cold leg piping. The leak detection system is capable of detecting the presence of steam/water leaks on the order of 0.45×10^{-5} kg/sec or larger and

of signaling within one to three minutes upon initiation of a leak, during normal operation. Operator action is taken upon receipt of a leak signal to shutdown the affected system, by closing steam/water isolation valves and depressurizing the affected unit.

- 379 (CONF-760735--2) DESIGN OF THE US-CRBRP SODIUM/WATER REACTION PRESSURE RELIEF SYSTEM. Kruger, G.B.; Murdock, T.B.; Rodwell, E.; Sane, J.D. (General Electric Co., Sunnyvale, CA (USA); Westinghouse Electric Corp., Pittsburgh, PA (USA)). 1976. Contract EY-76-C-15-2395. 24p. Dep. NTIS, PC A02/MF A01.

From US/USSR steam generator seminar; Dimitrovgrad, USSR (26 Jul 1976).

Protection against intermediate sodium system overpressure from the sodium/water reaction associated with large leaks within the CRBRP Steam Generators is provided by the sodium/water reaction pressure relief system (SWRPRS). This system consists of rupture disks connected to the intermediate sodium piping adjacent to the inlet to the superheater and outlet from the evaporator modules. The rupture discs relieve into piping that leads to reaction produce separator tanks, which in turn are vented to a centrifugal separator and flare stack arranged to burn hydrogen gas exhausting into the atmosphere. Analyses have been conducted using the TRANSWRAP Computer Code to predict the system pressures and flow rates during the large leak event. Experimental tests to be conducted in the large leak test rig (LLTR) will be used to confirm the analysis techniques used in the design.

- 380 (CONF-761001--3, pp 1047-1056) SAFETY PROBLEMS RELATED TO MICROHETEROGENEITIES IN PHYSICALLY MIXED OXIDE FUELS. Renard, A.; Evrard, G.; Vanhellemont, G. (Belgonucleaire SA, Brussels). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The safety aspects of microheterogeneities in LMFBR mixed oxide fuel are reviewed from the point of view of the pin behaviour dynamic study, the fabrication and the quality control. The paper emphasizes some significant parameters in transient conditions, the prevention means in the fabrication process and the analysis methods for the control.

- 381 (CONF-761001--P1, pp 2-13) KEY ISSUES ON SAFETY DESIGN BASIS SELECTION AND SAFETY ASSESSMENT. An, E.; Togo, Y. (Tokyo Univ.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

In current fast reactor design in Japan, four design accident conditions and four design seismic conditions are adopted as the design base classifications. These are classified by the considerations on both likelihood of occurrence and the severeness of the consequences. There are several major problem areas in safety design consideration such as core accident problems which include fuel sodium interaction, fuel failure propagation and residual decay heat removal, and decay heat removal systems problems which is more or less the problem of selection of appropriate system and of assurance of high reliability of the system. In view of licensing, two kinds of accidents are postulated in evaluating the adequacy of a reactor site. The one is the 'major accident' which is the accident to give most severe radiation hazard to the public from technical point of view. The other is the

"hypothetical accident", induced public accident of which is severer than that of major accident. While the concept of the former is rather unique to Japanese licensing, the latter is almost equivalent to design base hypothetical accident of the US practice. In this paper, design bases selections, key safety issues and some of the licensing considerations in Japan are described.

- 382 (CONF-761001--P1, pp 14-23) LICENSING ASPECTS IN THE VERIFICATION OF THE SNR 300 DESIGN CONCEPT AGAINST HYPOTHETICAL ACCIDENTS. Kugler, E.; Wiesner, S. (Ministerium fuer Arbeit, Gesundheit und Soziales des Landes Nordrhein-Westfalen, Duesseldorf). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The German prototype of a fast breeder reactor, the SNR 300, is being built near Kalkar on the Lower Rhine. It is a loop-type fast sodium-cooled reactor, designed and constructed by Interatom, Bensberg. Experiences gained from the first phase of construction are described. The report is restricted to the aspects of the SNR 300 design against a core disruptive accident (CDA) and its consequences and to the difficulties having arisen in the verification of the design concept so far. Some examples of the detailed design are described and discussed from the licensing authority's point of view showing that the difficulties have been typical for a prototype reactor subjected to a regular licensing procedure.

- 383 (CONF-761001--P1, pp 24-32) KEY SAFETY ISSUES FOR FAST BREEDER REACTORS. Tanguy, P.; Petit, J.; Justin, F. (Commissariat a l'Energie Atomique, Saclay, France). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Safety evaluations made on a commercial LMFBR have shown a number of key safety issues. Not one of them seems unsolvable, but studies and tests are still necessary, especially if it is thought desirable to simplify the system or increase the unit size of the power plant.

- 384 (CONF-761001--P1, pp 33-43) SAFETY ISSUES IN LICENSING COMMERCIAL LMFBR'S. Hannaford, J.; Fryer, D.R.H. (Nuclear Installations Inspectorate, London). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The paper describes briefly the role of the Nuclear Installations Inspectorate of the Health and Safety Executive as the independent licensing body within the United Kingdom and explains the part they play in the pre-licensing phases of commercial fast reactors. The whole core accident (WCA) is examined in some detail and from this examination the major safety issues are highlighted. The Inspectorate believes that all necessary safeguards should be provided to prevent the occurrence of WCAs, that the objective should be to maintain the core in a state which may be described by analytical models and for which appropriate data is available. Nevertheless, in view of the nature of whole-core accidents and of residual uncertainties associated with LMFBRs, it is considered necessary for designs to include suitable provisions for coping with WCAs. The importance is stressed of recognising that decisions made for the first commercial fast reactors will tend to commit the designs for subsequent power programs. Consideration must be given now, therefore, to ways of reducing

the yield from whole core accidents and of providing containments which can cope with upper bound yields. While research results may subsequently permit a reduction in the level of conservatism currently applied, safety decisions made now must be justifiable in the light of present knowledge.

- 385 (CONF-761001--P1, pp 44-51) SOME THOUGHTS ON FAST REACTOR SAFETY. Okrent, D. (Univ. of California, Los Angeles). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

This paper has been divided into five related sections: (1) ACRS recommendations on LMFBR safety; (2) Some historical perspectives on LWR safety, particularly as they relate to core melt; (3) Some speculation on societal approaches to low probability, high consequence accidents; (4) Hazards from LWR's and LMFBR's; and (5) Some personal thoughts on LMFBR safety.

- 386 (CONF-761001--P1, pp 61-75) SAFETY CONCLUSIONS FROM START-UP TESTS AND FROM THE ANALYSIS OF THE CORE BEHAVIOUR OF PHENIX. Caluzon, P.P. (Centre d'Etudes Nucleaires de Cadarache, Saint-Paul-Lex-Durance, France); Coulon, P.; Meyer-Heine, A.; Penet, F. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The start-up tests of PHENIX and the analysis of the core behavior for more than 2 years of operation have provided substantial information for the safety analysis of similar reactors. The validity of neutronic basic data and codes is confirmed by the good prediction of the main core physical characteristics, including the neutronic feedbacks which are more specifically involved in the accident consequence evaluation. Furthermore, sodium natural convection build-up has been proved experimentally, thus improving the knowledge on its essential influence on all loss of electric supply type accidents.

- 387 (CONF-761001--P1, pp 76-83) OPERATING EXPERIENCE WITH THE PHENIX NUCLEAR POWER STATION FROM THE POINT OF VIEW OF SAFETY. Goddet, J. (Phenix Plant, Bagnols-sur-Ceze, France). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Three years after the PHENIX first criticality, and after more than 3 billion kWh production, it is useful to present operating experience results regarding safety and protection. The main result concerns the sodium circuits radioactivity. It shows that where there is a risk of sodium leak or sodium-water reaction, there is no radioactivity. Radioactive and sodium risks are then separately examined. Waste in solid, liquid or gaseous form are specially described. Last is reported the behavior of some reactor components, the reliability of which has a direct influence on safety analysis.

- 388 (CONF-761001--P1, pp 84-98) SAFETY DESIGN OF SNR-300. Traube, K. (Interatom, Bensberg, France). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The joint German-Belgian-Dutch loop-type 300 MW(e) LMFBR prototype is being constructed at Kalkar on the lower Rhine in Germany. Among the many arguments put forward in defense of SNR-

300, that of acquiring licensing experience has proven to be of major importance to the international breeder scene. The severity of the licensing procedures and of the safety standards imposed are unique in several respects, including timing: generally growing scepticism towards nuclear power increased severity of the licensing practice; organizational features: the procedure and criteria developed for commercial light water reactors have been applied without exemptions. This relates to the commercial-type contract under which SNR-300 is being built for private utilities by a private company; and German nuclear safety standards, known worldwide to be most stringent. The following three important areas are discussed in which SNR-300 decidedly deviates from its forerunners: protection against the hypothetical core disruptive accident (HCDA), protection against external events, and provisions for in-service inspection.

- 389 (CONF-761001--P1, pp 99-111) SAFETY-RELATED DESIGN OF MONJU. Kawashima, K.; Suzuki, Y. (Power Reactor and Nuclear Fuel Development Corp., Tokyo). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

MONJU is a prototype reactor of 300 MWe. Construction of MONJU is scheduled to start in 1978, and it is now in the stage of preparing for license applications. Some of the safety design features of MONJU are: elevated main coolant pipes, guard vessels for reactor vessel, IHX's and pumps, three decay heat removal systems branched from IHX's, two independent safety rods, and cylindrical steel shell container. These designs are based on the safety design guidelines specified by PNC (Power Reactor and Nuclear Fuel Development Corp.). The guideline includes four design accident categories and four earthquake conditions. Core disruptive accident is chosen as the "hypothetical accident" for siting evaluation. In this paper, safety design guidelines and safety design features of MONJU are described.

- 390 (CONF-761001--P1, pp 112-128) SAFETY-RELATED DESIGN CONSIDERATIONS FOR THE CLINCH RIVER BREEDER REACTOR PLANT. Slember, R.J. (Westinghouse Electric Corp., Oak Ridge, TN). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

In this paper the engineered safety features and margins which have been incorporated into the design of the plant are described. The features emphasized are those to prevent accidents which could lead to core disruption, and to accommodate certain design requirements such as the site suitability source term. Also, described are the specific design margins and features which have the capability to mitigate the consequences of accidents beyond those included in the plant design basis, to provide additional assurance that the health and safety of the public are adequately protected.

- 391 (CONF-761001--P1, pp 164-179) MAIN SAFETY FEATURES OF THE SUPER-PHENIX PROJECT. Robert, E. (EDF/NERSA, Lyon); Lucet, G.; Chalot, A.; Leduc, J. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The construction of the CREYS-MALVILLE Fast Neutron Breeder Reactor (SUPER-PHENIX) started

in 1976 and industrial operation is scheduled for 1982. From the standpoint of safety the Plant presents some interesting features. For although the Plant repeats logically the same major options laid down for PHENIX, the international nature of the project, the size of the reactor (3000 MW thermal power), the new French licensing procedure to which it was subjected, have led the designers to provide a coherent aggregate of measures and/or devices aimed at ensuring the maximum intrinsic safety of the facilities. The purpose of this report is to describe and stress some of these features.

- 392 (CONF-761001--P1, pp 180-192) SEISMIC ANALYSIS FOR THE SUPER-PHENIX REACTOR. Lefevre, J.C. (CIRNA-PROJET SUPER-PHENIX, Gif-sur-Yvette, France); Livolant, M.; Lucet, G. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The seismic analysis performed for SUPER-PHENIX is presented in this paper. The analysis consists essentially of: selecting reference ground motions based on the seismic risks at the site; establishing general earthquake safety criteria; studying the response of buildings, systems and components, using the most appropriate methods; and establishing design and acceptance criteria in order to verify the validity of the conception and the reliability of the equipment.

- 393 (CONF-761001--P1, pp 193-208) SAFETY ANALYSIS ASPECTS OF A 2000 MWE LMFR CORE. Struwe, D. (Kernforschungszentrum, Karlsruhe, Ger.); Huebel, H.; Maschek, W.; Heusener, G.; Kieflhaber, E.; Kleefeldt, K. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

To identify areas of constraint possibly limiting the maximum allowable power of single plants safety analysis has been carried out for a 2000 MWe reactor core. Design characteristics of the SNR-2000 reactor are presented. Core disruptive accident analysis has been performed for transient overpower and loss of flow accidents. No basic difficulties have been found, in demonstrating that TOP cases are likely to lead to relatively mild excursions. Reference case evaluations of LOF accidents also showed no energetic power bursts. Due to uncertainties with regard to fuel and clad movement a parametric investigation of LOF initiated disassembly phases has been found to be necessary. Results demonstrated, that it is essential to limit the possibility of high positive reactivity ramp generation as long as mechanical restraints or thermal inertia of fuel are effective. Modifications of conventionally designed reactor cores are discussed. Despite the fact that uncertainties have been identified the study strongly supports that in principal conventionally designed large liquid metal fast breeder reactors may withstand consequences of core disruptive accidents.

- 394 (CONF-761001--P1, pp 209-223) SAFETY RELATED DESIGN CONSIDERATIONS FOR LARGE FAST BREEDER PLANTS. Griffith, J.D. (Energy Research and Development Administration, Washington, DC); Graham, J.; Greebler, P.; Lancet, R.T. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The safety and licensing philosophy for

large U.S. fast breeder plants is presented and the implementation of the safety approach in design is discussed. Highlighted in addition to the influence of the U.S. licensing process is the evolving use of risk analysis to provide a needed framework for systematically securing and organizing the information needed to make safety-related design decisions. The supporting safety development program is also discussed.

395 (CONF-761001--P1, pp 232-245)

CONSIDERATIONS ON A PAHR TEST FACILITY.

Boenisch, G.; Groetzbach, G.; Heinzel, V.; Kleeefeld, K.; Kuechle, M.; Mueller, R.A.; Roysl, P.; Schramm, K.; Smidt, D.; Werle, H. (Kernforschungszentrum, Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

On the basis of a hypothetical core disruptive accident (HCDA) analysis the phenomena of the post accident phase are first identified which require experimental investigations and can only be studied in pile. Then the experimental requests for both debris bed and molten fuel pool studies are specified and grouped into three categories. For two of the categories the requests can be satisfied with loop experiments in thermal reactors. For the third category a 70 cm diameter test bed is needed and here the proposal is to use a flat core fast reactor with the test bed located below the core heated by axial leakage neutrons. Finally a conceptual design for such a reactor is presented where the test bed is loaded into an ex-vessel device and is removable on a carriage to a hot cell building. Maintenance and safety problems are briefly discussed and alternative solutions are mentioned.

396 (CONF-761001--P1, pp 246-261) THE SAREF PROGRAM. Avery, R. (Argonne National Laboratory, IL); Millunzi, A.C.; Grolmes, M.A.; Rose, D. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The current proposed SAREF program for acquiring the necessary facilities and performing the in-pile experiments required to resolve the key issues in fast reactor safety are reviewed. The facility acquisition plans involve both modification of existing facilities, TREAT and EBR-II, and the acquisition of a major new facility, the Safety Test Facility (STF). The proposed characteristics of the facilities are discussed, as are the considerations that led to the decisions made on facility selection.

397 (CONF-761001--P1, pp 262-269) SAFETY TEST FACILITIES. NEEDS AND CONCEPTS. A FRENCH EVALUATION. Tretjakoff, O.; Bailly, J. (Commissariat a l'Energie Atomique, Saclay, France). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The fuel behavior of LMFBRs in the event of an accident has been tested in-pile in the SCARABEE program. These tests will be carried on in the framework of an international cooperation on irradiated fuels: this is the purpose of the CABRI and SCARABEE N programs. All those studies should enable to assess safety margins between accident conditions and the technical specifications of the reactor. The purpose of this paper is to explain how a logical set of simple observations has led us

to the present state of the Cadarache in-pile experimental safety program and how it may help us to find our way in a dense forest of both technical and psychological difficulties.

398 (CONF-761001--P2, pp 512-520) EXTRAPOLATION OF ZPR SODIUM VOID MEASUREMENTS TO THE POWER REACTOR. Beck, C.L.; Collins, P.J.; Lineberry, M.J.; Grasseschi, G.L. (Argonne National Lab., Idaho Falls, ID). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Sodium-voiding measurements of ZPPR assemblies 2 and 6 are analyzed with ENDF/B Version IV data. Computations include directional diffusion coefficients to account for streaming effects resulting from the plate structure of the critical assembly. Bias factors for extrapolating critical assembly data to the CRBR design are derived from the results of this analysis.

399 (CONF-761001--P2, pp 555-564) SAFETY ASPECTS IN NUCLEAR CORE DESIGN OF LMFBRs. Wehmann, U. (INTERATOM, Bensberg, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Nuclear investigations were performed in view of an improvement of the safety characteristics of SNR 2. The Doppler effect, the sodium void effect and burnup behaviour of cores with a heterogeneous arrangement of fuel and blanket elements together with the effect of moderation of both element types were investigated. Numerical results of parametric studies and of detailed calculations on some selected systems are presented.

400 (CONF-761001--P2, pp 565-574) LOW SODIUM VOID COEFFICIENT LMFBR CORES. Sehgal, B.R.; Lin, C.; Naser, J.; Loewenstein, W.B. (Electric Power Research Inst., Palo Alto, CA). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A major safety concern in the design of commercial-sized LMFBR's has been the addition of positive reactivity with coolant voiding. Two different approaches to alleviate this problem are discussed. The first is a core design using modular assemblies with $\text{PuO}_2\text{-UO}_2$ fuel. Neutronic and fuel cycle calculations show that the positive coolant voiding Δk may be decreased substantially without incurring significant penalties in the fissile doubling time or the Doppler coefficient. This design does require a larger initial fissile inventory. The second approach is to use thorium-based fuels in uniform cores. Neutronic and fuel cycle analyses show that thorium metal alloys may be attractive candidates for LMFBR fuels. They appear to provide higher breeding gains, lower positive coolant void reactivity changes, higher negative Doppler coefficients, slightly lower fissile inventories and longer fuel cycle lengths than the reference $\text{PuO}_2\text{-UO}_2$ fuel. The $\text{PuO}_2\text{-ThO}_2$ fuel also shows favorable safety parameters compared to the reference core.

401 (CONF-761001--P2, pp 575-583) SAFETY AND BREEDING: RELATED ASPECTS OF FAST REACTOR CORES HAVING INTERNAL BLANKETS. Driscoll, M.J.; Ducat, G.A.; Pinnock, R.A.; Aldrich, D.C. (Massachusetts Inst. of Tech., Cambridge).

1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The safety characteristics and breeding performance of fast reactor cores having internal axial blankets are examined. Worthwhile improvements and tolerable penalties are identified in both areas based on comparisons with conventional core designs having identical external and fuel assembly dimensions. Internally-blanketed cores have smaller sodium void reactivity contributions and, if properly designed, higher breeding gains and shorter fissile inventory doubling times than conventional cores. The potential for additional improvements in breeding gain is also identified for internally-blanketed cores which are optimized to exploit the inherently lower neutron fluence and fluence/power/temperature gradients characteristic of these cores.

402 (CONF-761001--P2, pp 584-593)

SIMULATION OF AN HCDA SEQUENCE ON THE ZPPR CRITICAL FACILITY. Kaiser, R.E.; Beck, C.L.; Lineberry, M.J. (Argonne National Lab., Idaho Falls, ID). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

ZPPR Assembly 5 was used to simulate successive stages in a loss-of-flow initiated Hypothetical Core Disruptive Accident: sodium-voiding, clad redistributions, and fuel redistribution. Each stage was subdivided into steps corresponding to the anticipated accident sequence. The reactivity change in each step was measured, and reaction rate and small-sample reactivity distributions were measured in each of the three accident stages. The measurements were adequately predicted using diffusion theory, with some transport and streaming corrections.

403 (CONF-761001--P2, pp 594-601) ANALYSIS OF SIMULATED HCDA EXPERIMENTS IN ZPPR.

Stewart, S.L.; Lee, B.W.; Hitchcock, J.T.; Crick, S.C. (General Electric Co., Sunnyvale, CA). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

An experimental program was run in the ZPPR critical facility in order to evaluate the design methods used to provide the nuclear feedback information for the hypothetical core disassembly accident analysis. A series of "snapshot" views of a progressing accident was mocked up in the reactor and measurements were made of the resulting reactivity. The accident scenario consisted of voiding sodium, followed by slumping 30 percent of the steel, followed by slumping fuel both toward and away from the core center. Analysis of these experiments indicates that corrections for neutron streaming are necessary to properly analyze data from plate critical facilities. However, if measured worths are adjusted to convert them to equivalent pin data, perturbation theory is an adequate tool for predicting the reactivity effect of sodium voiding and steel redistribution. Diffusion theory was found to be inadequate for accurately predicting the reactivity worth of extreme conditions of fuel redistribution.

404 (CONF-761001--P2, pp 605-614) SAFETY PHILOSOPHY IN UPGRADING THE EBR-II PLANT PROTECTION SYSTEM. Sackett, J.I. (Argonne National Lab., Idaho Falls, ID). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The EBR-II plant protection system (PPS) has been substantially modified, upgrading its performance to more fully comply with modern safety philosophy and criteria. The upgrading effort required that the total reactor system be evaluated for possible faults and that a PPS be designed to accommodate them. The result was deletion of a number of existing trip functions and upgrading of others. Particular attention was given to loss of primary pumping power and reactivity insertion events. The design and performance criteria for the PPS has been more firmly established, understanding of the PPS function has been improved and the reactor has been subjected to fewer spurious trips, improving operational reliability.

405 (CONF-761001--P2, pp 615-624) SAFETY RELATED CONSIDERATIONS FOR OPERATION WITH DEFECTED ELEMENTS IN EBR-II. Fryer, R.M.; Sackett, J.I.; Lambert, J.D.B. (Argonne National Lab., Idaho Falls, ID). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Traditionally, EBR-II has employed the 'shutdown and remove' philosophy when breached fuel elements are encountered. This mode of operation maintained in-plant inventories of fission products at low levels and allowed certain fission product detection systems to be employed as automatic plant shutdown devices. Information from fuel failure propagation studies and fast reactor operation indicates that shutdown under these conditions is unwarranted. Analytical studies, as well as fast reactor experience, further indicate that failure propagation, if it occurs at all, will not cross adjacent subassembly boundaries. Therefore, the 'shutdown and remove' philosophy can be liberalized to allow the demonstration of safety during a run-beyond-clad-breach mode of operation. This mode of operation is essential to the demonstration of the economics of commercial LMFBR systems.

406 (CONF-761001--P2, pp 625-634) DETECTION OF COOLANT BLOCKAGES: PROBLEMS AND POSSIBILITIES. Greef, C.P.; Clare, A.J.; Board, S.J.; Hall, R.S. (Central Electricity Generating Board, Berkeley, Eng.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A semi-empirical model is developed correlating a wide range of experimental data on peak wake temperatures for subassembly blockages; it is predicted that boiling could be produced at a corner blockage of 10 percent flow area in a 325 pin subassembly. An assessment of the possible consequences of local boiling suggests that the margin to propagation may be small, particularly in the presence of fission gas release, leading to a requirement for early detection systems. A continuing theoretical and experimental programme to investigate the temperature noise technique for blockage detection is reported. It is concluded that blockages appreciably smaller than those predicted to produce boiling should be detectable against a background noise level due to subassembly power tilts; considerable statistical certainty would be obtained with a 5 second detection time, giving protection against rapidly developing incidents. The use of wire wrapped fuel pins would raise the detection threshold by a factor approximately 2 to 3 in blockage area.

- 407 (CONF-761001--P2, pp 635-645) DESIGNS FOR IN-REACTOR SODIUM BOILING DETECTION AND GENERATION. Wright, S.A. (Sandia Labs., Albuquerque, NM); Albrecht, R.W.; Artus, S.C.; Crowe, R.D.; Dodd, L.R. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Design of a detection system and sodium boiling generation devices is presented. It is shown that if the probability of a sodium boiling incident is sufficiently high, then a detection system is economically feasible. It is also shown that safe and reliable sodium boiling generation devices can be operated in EBR-II.

- 408 (CONF-761001--P2, pp 658-664) LMFBR: SUBASSEMBLY FAULT DETECTION AND IDENTIFICATION BY TEMPERATURE NOISE ANALYSIS. Ohlmer, E.; Schwalm, D. (EURATOM, Ispra, Italy). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The aim of these investigations is to check if it would be possible in principle to detect and to identify a malfunction in a LMFBR-subassembly (e.g. blockages) by analysing the temperature noise measured downstream from the bundle exit. Temperature noise measurements have been performed in different test sections simulating the thermohydraulic conditions downstream from the fuel bundle exit using water as a simulant fluid. As a result it was found by analysing the amplitude distribution of the temperature fluctuations that it is possible to identify the size of different simulated blockages. However, in practice the analyzing techniques developed can be applied successfully only if the scale of turbulence is sufficiently large and if no mixer will destroy the relevant information in the zone between the bundle exit and the temperature fluctuation detector.

- 409 (CONF-761001--P2, pp 665-672) INDICATION OF A COOLANT BLOCKAGE IN A FUEL ELEMENT OF A SODIUM-COOLED REACTOR BY TEMPERATURE FLUCTUATION MEASUREMENTS WITH STEEL-SODIUM THERMOCOUPLES. Krebs, L.; Weinkoetz, G. (Kernforschungszentrum, Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The knowledge of the coolant temperature fluctuations measured at the outlet of a subassembly is an important indicator for the control and safety of sodium cooled reactors. For this reason measurements were carried out at the fuel element outlet of the KNK I reactor and on an electrically heated 28-rod bundle. This bundle has a 60 degree segment geometry and a blocked section of 68, 5 percent. On both test facilities the root mean square (RMS) and power spectral density (PSD) of temperature signals from intrinsic steel-sodium thermocouples were analyzed. Moreover the influence of a flow mixer was investigated on the 28-rod bundle. The results of these measurements show that the PSD and the RMS analyses seem to be significant sizes to detect flow blockages in reactors, possibly behind flow mixers.

- 410 (CONF-761001--P2, pp 683-695) HYDRAULICALLY SUPPORTED ABSORBER BALLS SHUTDOWN SYSTEM FOR INHERENTLY SAFE LMFBR'S. Specht, E.R.; Paschall, R.K.; Marquette, M.; Jackola, A. (Atoms International, Canoga Park, CA).

1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A diverse, inherently-actuated control system would be very valuable for Advanced LMFBR's since its use may preclude the consideration of a "hypothetical core disruptive accident" as a design basis. An Inherent Shutdown System (ISS) which uses a number of subassemblies, each containing a column of hydraulically supported tantalum balls, is being studied. This system appears to be capable of successfully protecting the public (and the reactor) from low-probability high-consequence events, such as a loss-of-flow or transient-over-power incident with failure of the normal control system (or systems) to scram. A number of columns of small (about 1/4 in. OD) tantalum balls are held above the active core region of a reactor by the reactor coolant flow. These columns automatically fall into the core and shut the reactor down if the coolant flow of the reactor is interrupted by a loss-of-flow event. A thermally-actuated valve within each ISS subassembly enables this system to shut down the reactor during a transient-over-power event. This control system is extremely diverse since it can operate in a severely distorted core and is independent of top shield rotation. Also, it is basically very simple and should be extremely reliable after it is developed.

- 411 (CONF-761001--P2, pp 709-718) SAFETY EVALUATION OF THE MONJU STEAM GENERATOR. Hori, M.; Kawashima, K.; Sato, M.; Saito, T.; Kanegae, N.; Hiroi, H. (Power Reactor and Nuclear Fuel Development Corp., Tokyo). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

For the design and safety evaluation of MONJU steam generator, the safety design guides are determined and the models and input data for the sodium-water reaction analysis are specified. These guides and specification are being updated and verified with the progress of the design, analysis and tests. Safety evaluation at the current stage of MONJU design shows that the steam generator and the secondary circuit components could withstand, with sufficient margin, the pressure effect of the maximum conceivable leak.

- 412 (CONF-761001--P2, pp 721-727) RELIABILITY MODELS FOR PLANT PROTECTION SYSTEMS. Hockenbury, R.W.; Yeater, M.L. (Rensselaer Polytechnic Inst., Troy, NY). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Methods to calculate system reliability and the confidence levels of such calculations are being evaluated and modified on the basis of their usefulness in operating plants, and the CRBR in particular. Model development from this viewpoint includes study of approximate methods and the determination of their accuracy by comparison with benchmark calculations and Monte Carlo simulation. As a part of this approach to operational failure prediction and maintenance scheduling, the role of sensor and channel response in the partial failure regime is considered.

- 413 (CONF-761001--P2, pp 748-756) ACCIDENT TREES: A FRAMEWORK FOR ACCIDENT EVALUATION. Bradfute, J.D.; Gilchrist, A.R.; Mott, J.E. (General Electric Co., Sunnyvale, CA). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The event tree formalism is used in a data base system for LMFBR safety-related information. The central part of this system is a quantified data base explicitly dependent upon individual event tree branch points. A formalism for including the effects of proposed experiments, test facilities or design options is developed. The output of the system is sets of data which form input for decision analyses. The feasibility of the approach is demonstrated by applying simplified data from previous studies of safety research facility informational needs, costs, and benefits. We conclude that such a system has wide application to organizing, clarifying and forming decision bases for LMFBR safety-related information.

- 414 (CONF-761001--P2, pp 757-764) COMPREHENSIVE METHOD OF COMMON-MODE FAILURE ANALYSIS FOR LMFBR SAFETY SYSTEMS. Unione, A.J.; Ritzman, R.L.; Erdmann, R.C. (Science Applications, Inc., Palo Alto, CA). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A technique is demonstrated which allows the systematic treatment of common-mode failures of safety system performance. The technique uses log analysis in the form of fault and success trees to qualitatively assess the sources of common-mode failure and quantitatively estimate the contribution to the overall risk of system failure. The analysis is applied to the secondary control rod system of an early sized LMFBR.

- 415 (CONF-761001--P2, pp 787-796) IN-PILE EXPERIMENTS AND TEST FACILITIES PROPOSED FOR FAST REACTOR SAFETY. Grohmes, M.A.; Avery, R.; Goldman, A.J.; Fauske, H.K.; Marchaterre, J.F.; Rose, D.; Wright, A.E. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The role of in-pile experiments in support of the resolution of fast breeder reactor safety and licensing issues has been re-examined, with emphasis on key safety issues. Experiment needs have been related to the specific characteristics of these safety issues and to realistic requirements for additional test facility capabilities which can be achieved and utilized within the next ten years. It is found that those safety issues related to the energetics of core disruptive accidents have the largest impact on new facility requirements. However, utilization of existing facilities with modifications can provide for a continuing increase in experiment capability and experiment results on a timely basis. Emphasis has been placed upon maximum utilization of existing facilities and minimum requirements for new facilities. This evaluation has concluded that a new Safety Test Facility, STF, along with major modifications to the EBR II facility, improvement in TREAT capabilities, the existing Sodium Loop Safety Facility and corresponding Support Facilities provide the essential elements of the Safety Research Experiment Facilities (SAREF) required for resolution of key issues.

- 416 (CONF-761001--P2, pp 846-855) FUEL AND CLADDING MOTION SURVEILLANCE. De Volpi, A. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor

safety and related physics; Chicago, IL, USA (5 Oct 1976).

The central role of fuel and clad motion is examined from the viewpoint of diagnostic instrumentation. Direct surveillance of material motion during tests conducted in safety test facilities is necessary to provide various postulated models with realistic chronological data. Other roles for such surveillance are identified. The fast neutron hodoscope at TREAT has demonstrated its ability to satisfy experiment requirements, and some results are highlighted. Improvements and extensions of surveillance are described, and plans to fulfill a multi-facility program for new or modified safety facilities are discussed, with emphasis on the need for adequate sensitivity to material motion in large test sections.

- 417 (CONF-761001--P2, pp 864-873) CODED APERTURE IMAGING METHODS FOR FUEL MOTION MEASUREMENT. Kelly, J.G.; Stalker, K.T. (Sandia Labs., Albuquerque, NM). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Coded aperture imaging techniques are being applied to fuel motion detection in core disruptive safety experiments in the LMFBR. Its potential for time resolved broad source X-radiography is being investigated, and for the Sandia Annular Core Pulse Reactor (ACPR) a system is being developed to image moving fuel with approximately 1 mm spatial resolution at about 10^4 frames/sec. Thus far in the investigations of these systems a number of feasibility milestones have been reached. (1) A fuel pin shaped object has been imaged with 2 mm resolution using a large area 10 MeV electron bremsstrahlung spectrum and coded source X-radiography. (2) An extended ^{252}Cf fission gamma ray source has been imaged with film through 1.27 cm of steel (simulating reactor containment). (3) The ^{252}Cf extended source has been imaged through 1.27 cm of steel with X-ray image intensifier amplification. (4) Collimation and shielding tests indicate that at ACPR it should be possible to obtain adequate signal to noise at the detector when the fuel is observed through a slot in the core.

- 418 (CONF-761001--P2, pp 492-501) RECENT IMPROVEMENTS IN THE PREDICTION OF REACTIVITY COEFFICIENTS RELEVANT TO LMFBR ACCIDENT ANALYSIS. Fischer, E.A.; Helm, F. (Kernforschungszentrum, Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The results of measurements of the effective delayed neutron fraction β_{eff} are given for various assemblies built in the fast critical facility SNEAK. They are in good agreement with a recent evaluation of delayed neutron data by Tuttle. β_{eff} values derived from the Tuttle data also give an improved agreement between measurements and calculations for the material worths of fissile isotopes. Further, measurements and calculations of the sodium void effect in assembly 9C of SNEAK are described in detail. While the agreement for oxide compositions using standard SNEAK fuel is satisfactory, the effect of introducing higher Pu-isotopes and of using carbide instead of oxide composition causes calculational difficulties.

- 419 (CONF-761001--P2, pp 728-737)

APPLICABILITY OF THE REACTOR SAFETY STUDY (WASH-1400) TO LMFBR RISK ASSESSMENTS. El-Sheikh, K.A. (General Electric Co., Sunnyvale, CA); Feller, K.G.; Fleischer, L.; Greebler, P.; McDonald, A.; Sultan, P.; Temme, M.I.; Fullwood, R.R. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The feasibility of applying the WASH-1400 methods and data to LMFBR risk assessment is evaluated using the following approach for a selected LMFBR: (1) Structuring the LMFBR risk assessment problem in a modular form similar to WASH-1400; (2) Comparing the predictive tools applicable to each module; (3) Comparing the dependencies among the various modules. It is concluded that the WASH-1400 applicability is limited due to LWR-LMFBR differences in operating environments and accident phenomena. WASH-1400 and LMFBR specific methods applicable to LMFBR risk assessments are indicated.

- 420 (CONF-761001--P2, pp 369-387) LMFBR ACCIDENT ENERGETICS AND THEIR ROLE IN LICENSING. Speis, T.P. (Nuclear Regulatory Commission, Washington, DC); Denise, R.P.; Starostek, R.W.; Lois, L.; Meyer, J.F.; Theofanous, T.G. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The role of core disruptive accidents in the licensing of LMFBR's is addressed, in particular as it relates to the Clinch River Breeder Reactor. The evolution of the Nuclear Regulatory Commission's containment design requirements is discussed and the technical basis for its present position reviewed. The initiators considered for energetic core disruption are loss-of-flow (LOF) and transient overpower (TOP) both without scram. In addition to considering the immediate mechanical disassembly potential from these initiators, the potential for recriticality during meltdown phases is also assessed. The loss-of-flow cases are given particular attention because of the apparent potential for significant energetics.

- 421 (CONF-761001--P2, pp 400-410) RESOLUTION OF KEY SAFETY-RELATED ISSUES IN FFTF REGULATORY REVIEW. Simpson, D.E. (Westinghouse Hanford Co., Richland, WA). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The FFTF is an ERDA facility which does not require licensing, but a technical review by NRC is required by ERDA policy. Safety issues which were not fully resolved in the course of the review for construction authorization have been the subject of continuing review since mid-1973. These issues included HCDA energetics, design fallback provisions for additional safety margins, piping integrity, and natural circulation core cooling.

- 422 (CONF-761001--P2, pp 411-419) DEVELOPMENT OF TECHNICAL SPECIFICATIONS FOR EBR-II: SOME CONSIDERATIONS FOR APPLICATION TO COMMERCIAL LMFBR'S. Sackett, J.I.; Gale, N.L. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Considerations made in development of Technical Specifications for EBR-II identified operational-safety concerns unique to LMFBR's and applicable to both major reactor test facilities and prototype commercial plants.

This effort highlighted the fact that to be useful, such documents must be prepared with a clear understanding of purpose and application. The necessary difference in purpose of these documents for different reactor applications is not widely recognized. For commercial reactors, the form and application of these documents has gone through significant evaluation. The Technical Specifications developed for EBR-II define the safe operating envelope while at the same time providing the flexibility to accommodate a wide range of tests relevant to fast reactor safety. This required, in many cases, definition of performance criteria rather than specific operating limits. Much consideration was given to differentiating between items truly relevant to personnel safety and those relevant only to reactor availability.

- 423 (CONF-761001--P2, pp 420-430) DESIGNING FOR SAFETY IN FAST REACTORS IN THE PRESENCE OF UNCERTAINTIES. Cave, L.; Ilberg, D.; Okrent, D. (Univ. of California, Los Angeles). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Topics discussed include safety requirements for fast reactors, sources of uncertainty in the demonstration of high reliability requirement in reactor systems, and possible means of limiting the risk from pressure-driven core disassembly accidents.

- 424 (CONF-761001--P2, pp 442-451) DECAY HEAT REMOVAL PLAN OF THE SNR-300: A LICENSED CONCEPT. Morgenstern, F.H.; Gyr, W.; Stoetzel, H.; Vossebrecker, H. (INTERATOM, Bensberg, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The report describes how the decay heat removal plan of the SNR-300 has been established in 3 essential licensing steps, thus giving a very significant example for the slow but steady progress in the overall licensing process of the plant. (1) Introduction of an ECCS in addition to the 3 main heat transfer chains as a back-up for rather unlikely and undefined occurrences, 1970; (2) Experimental and computational demonstration of a reliable functioning of the in-vessel natural convection of the fluid flow, 1974; and (3) Proof of fulfilling the general safety and specific reliability criteria for the overall decay heat removal plan; i.e., the 3 main heat transfer chains with specific installations on the steam/water system side and the ECCS, 1976. Some special problem areas, for instance the cavity concept provided for the pipe fracture accident, have still to be licensed, but they do not contribute considerably to the overall risk.

- 425 (CONF-761001--P2, pp 452-459) DESIGN REQUIREMENTS FOR THE SNR-300 CONTAINMENT SYSTEM. Deynhausen, H.; Morgenstern, F.; Scholle, U.; Lange, L.; Waldhoer, G. (INTERATOM, Bensberg, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The essential design data of the SNR-300 containment system were derived from the results of analyses of the Bethe-Tait-Accident (hypothetical core disruption). The thermodynamic loads to be expected after a Bethe-Tait-excursion are carried by the inner containment. The confinement of the activity

released following a Bethe-Tait-excursion is achieved by use of a reventing system in the outer containment. The essential results of the thermodynamic analysis are given as well as the design data of the system. The results of the aerosol-physical analysis are discussed with regard to their importance for the safety of the auxiliary equipment. The radiological considerations following an accident place special requirements upon the containment design. The radiological safety of the equipment is shown, even in the extreme case when active heat sinks are neglected.

- 426 (CONF-761001--P2, pp 463-472) COMPARISON OF NEUTRONICS CALCULATIONS IN A LOSS-OF-FLOW ACCIDENT. Schaefer, R.W. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Results of a study of the accuracy of point kinetics reactivity predictions for loss-of-flow transients in a CRBR model are presented. Neutronics calculations were driven by realistic material and temperature histories representative of the thermal-hydraulic feedback which occurs in this type of transient. Reactivity solutions from 3 point kinetics variations and the quasistatic method are compared. One variation is found to be superior to the others for these transients; its prediction at power burst conditions is low by approximately \$0.35. Accurate quasistatic neutronics calculations are found to have small CPU requirements relative to the time required for thermal-hydraulic calculations with SAS.

- 427 (CONF-761001--P2, pp 473-481) ASSESSMENT OF TRANSPORT EFFECTS IN LMFBR SAFETY NEUTRONICS. Cahalan, J.E. (Purdue Univ., Lafayette, IN); Ott, K.O.; Ferguson, D.R. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A qualitative and quantitative assessment of the significance of neutron transport effects in LMFBR core disruptive accident analysis is presented. Material relocations which might cause important neutron transport behavior are identified. A quantitative measure of the error in the neutron flux is obtained from a consistent numerical comparison of transport and diffusion theory eigenvalue solutions for models of disrupted cores. A numerical technique for the prediction of transport eigenvalues and eigenvectors is formulated and applied. The technique is based on a modified diffusion theory which is fully capable of reproducing transport theory solutions.

- 428 (CONF-761001--P2, pp 482-491) NEUTRONIC COUPLING OF FEEDBACK EFFECTS. Malloy, D.J.; Ott, K.O. (Purdue Univ., Lafayette, IN). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The currently applied feedback methodology has been extended by explicitly treating the coupling effect which exists between the various feedback components and by improving the description of the temperature dependence of the Doppler feedback. The Doppler and sodium void feedback and coupling coefficients are presented as a first comparison of the improved and the currently applied methods. Further, the energy release which results from hypothetical disassembly accidents is simulated with a special response surface (based upon VENUS-II

results) in the parametric safety evaluation code PARSEC. Additionally, the probability distribution for available work-energy, which results from the statistical uncertainty of the Doppler coefficient, is calculated. The improved feedback methodology yields about a 16 percent higher average value for the work-energy. A substantially larger increase is found on the high-yield end of the spectrum: the probability for work-energy above 500 MJ is increased by about a factor of ten.

- 429 (CONF-761001--P3, pp 885-894) METHOD FOR ASSESSING CORE-WIDE INCOHERENCY EFFECTS ON LMFBR WHOLE-CORE ACCIDENT EVOLUTION. Massey, J.V.; Lois, L.; Speis, T.P.; Theofanous, T.G. (Purdue Univ., Lafayette, IN). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A technique for the evaluation of the effects of core-wide incoherencies on the CRBRP unprotected Loss-of-Flow accident scenario has been presented. It appears that core-wide incoherencies result in increased core voiding and hence less sodium reactivity inventory, at the inception of LOP=d-TOP conditions. The approach to this condition is also governed by substantially lower ramp rates. Finally the post failure phenomena when treated with the degree of incoherency physically present, yield substantially milder super prompt critical conditions.

- 430 (CONF-761001--P3, pp 1037-1046) BEHAVIOR OF CARBIDE FUEL IN LOCAL AND WHOLE CORE ACCIDENTS. Siegmann, E.R.; Bergeron, R.J.; Nickerson, G.M. (Combustion Engineering, Inc., Windsor, CT). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

This paper evaluated the likelihood of carbide fuel pin failure propagation and describes the behavior of carbide fuel during whole core accident sequences, such as transient overpower (TOP) and transient undercooling (TUC). Applicable methodology is used to assess the possibility that postulated local accidents will result in the propagation of pin failure across a carbide subassembly. Of the three propagation mechanisms considered, local blockages, fuel coolant interaction (FCI), and fission gas release, only FCI is judged to have more than a remote possibility of causing propagation. Comparison is made between oxide and carbide fuel pins during both TOP and TUC. Results indicate that the sequence of events can vary with fuel type. Sodium bonding is found to influence the onset of cladding motion.

- 431 (CONF-761001--P3, pp 877-884) CEC ACTIVITIES IN THE FIELD OF LMFBR SAFETY. Balz, W.; Finzi, S.; Klersy, R. (Joint Research Centre, Ispra, Italy). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The aim of the ECC is to reach a common LMFBR Safety strategy in Europe. To this end the Commission promotes collaboration between the different fast reactor projects in the Community through working groups and collaborative arrangements and contributes with a research activity executed in its Joint Research Centre Ispra. A short description is given of the activity in the working groups and of the Ispra programme on LMFBR Safety. This programme covers: LMFBR thermohydraulics, fuel coolant interactions, dynamic structure loading

and response, safety related material properties and whole core accident code development.

- 432 (CONF-761001--P3, pp 905-914) ASSESSMENT OF FUEL FAILURE PROPAGATION IN LMFBRs. Fauske, H.K.; Grolmes, M.A.; Chan, S.H. (Argonne National Lab., IL). 1976.
- From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).
- The potential for subassembly disruption as well as the consequences thereof have been examined for an oxide-fueled liquid metal cooled fast breeder reactor. Based upon credible initiators, it has not been possible to identify any sequence of events which could result in subassembly disruption prior to detection and shutdown of the reactor. The principal reasons for this conclusion are: (1) local faults have been previously shown to be isolated events and; (2) slow blockage propagation appears non-mechanistic. Furthermore, even if subassembly meltdown is postulated to occur in the absence of scram, it was concluded that significant damage beyond the incident subassembly also appears unlikely. Finally, even if extensive whole-core involvement is postulated, the level of accident energetics is assessed to be insufficient to breach the primary containment. The principal reasons for the "benign" character of the consequences following postulated fuel disruption are attributed to the inherent dispersive nature of the heat generating fuel and the lack of satisfying the necessary requirements for large mass vapor explosions with the oxide-fuel-sodium system. These characteristics can be used in a general way to rule out early mechanical damage to the core and significant propagation beyond the incident subassembly due to radial meltthrough, as well as prompt critical conditions due to sodium voiding and recriticality events.

- 433 (CONF-761001--P3, pp 926-935) COMPUTATIONAL ANALYSIS OF FUEL--SODIUM INTERACTIONS WITH AN IMPROVED METHOD. Jacobs, H. (Kernforschungszentrum, Karlsruhe, Ger.). 1976.
- From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).
- A new fuel-sodium interaction model programmed as the MURTI code is presented. Owing to a more rigorous treatment of the heat fluxes within fuel and sodium in connection with a space and time-dependent description of the coolant state, it is more generally applicable than the models used so far. Furthermore in this model the region of interaction may be subdivided which enables a very flexible description of thermal interactions. It is demonstrated that most of the presently used models adequately describe only interactions with strong intermixing of fuel and sodium. In a case with only poor intermixing MURTI predicts early vaporization and the formation of a vaporous layer at the fuel surface while the remaining coolant acts as a heat sink only. Furthermore the MURTI code is applied to study thermal detonations as hypothesized by Board and Hall. It is found that an interaction with pressures far below the level of stable propagation can escalate by itself, if appropriate conditions are encountered.

- 434 (CONF-761001--P3, pp 947-957) EFFECT OF FAILURE INCOHERENCE IN A LMFBR HYPOTHETICAL OVERPOWER ACCIDENT. Royl, P.; Kuczera, B.

(Kernforschungszentrum, Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Incoherency effects due to the statistical nature of fuel rod failure have been assessed for transient overpower (TOP) accidents with different initiating ramp rates. Characteristic single subassembly feedbacks from sodium voiding, fuel injection, and fuel sweep out after failure were calculated and superimposed incoherently. Effective feedback functions were thus generated for characteristic subassembly groups (channels) and used as tabular reactivity input to corresponding multi-channel whole core TOP simulations using the HOPE computer model. Analyses were performed for the SNR-300 MARK 1A end of life (EOL) core with initiating ramp rates of 0.15, 0.5 and 5 \$/sec. The results are compared with those from a calculation with coherent subassembly behavior in each channel. It is concluded for the mild transient: Irrespective of the assumed axial failure location, negative fuel sweep out reactivity from early failed subassemblies will prevent the accident from reaching a prompt critical stage and lead to a gradual shut down, if the assessed failure incoherencies are applied. For the higher ramp rates (greater than or equal to 0.5 \$/sec) the assessed incoherencies will not always lead to an early shut down, but they can considerably reduce the potential to accumulate high ramp rates in the initiating phase of TOP accidents.

- 435 (CONF-761001--P3, pp 958-968) ANALYSIS OF LMFBR OVERPOWER ACCIDENTS, INCLUDING FISSION GAS EFFECTS IN THE PREDISASSEMBLY AND DISASSEMBLY PHASE. Bogenberger, H.G.; Fischer, E.A.; Royl, P.; Arnecke, G. (Kernforschungszentrum, Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The analysis of unprotected TOP accidents for an irradiated LMFBR core with the codes HOPE and KADIS was improved in two different ways: First, an existing fission gas behavior model was further developed to include a simulation of transient gas release. This model was used to check, and to improve the analysis of the gas behavior during steady state, and also in the predisassembly and disassembly phase of the accident. The effect of transient gas release before melting, which is predicted for a mild TOP, has been simulated in HOPE by adjusting the corresponding gas release parameters. In KADIS, a new equation of state, including fission gas effects, was introduced. Case studies were carried out for the LOL core of the SNR300. For accidents with different ramp rates, pin failure was assumed to occur in the axial peak node, at the onset of unrestructured fuel melting. These cases all lead to energetic core disassemblies and demonstrate the shutdown potential of the fission gas pressure under rather pessimistic conditions. In addition, the influence of transient gas release below the solidus was studied, in connection with a mechanical failure criterion. With this more realistic criterion, the accident is much milder. Gas release below the solidus did not greatly change the overall accident sequence; especially, it did not lead to more coherent failure of the channels, which would be associated with higher energy release.

- 436 (CONF-761001--P3, pp 979-987) EPIC: A COMPUTER PROGRAM FOR FUEL-COOLANT INTERACTIONS.

Pizzica, P.A.; Abramson, P.B. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

EPIC models fuel and coolant motions which result from internal fuel pin pressure (from fission gas or fuel vapor) and/or from the generation of sodium vapor pressures in the coolant channel subsequent to pin failure in an LMFBR. The modeling includes the ejection of molten fuel from the pin into a coolant channel with any amount of voiding through a clad rip which may be of any length or may expand with time. One-dimensional Eulerian hydrodynamics is used to model both the motion of fuel and fission gas inside a molten fuel cavity and the mixture of two-phase sodium and fission gas in the channel. Motion of molten fuel particles in the coolant channel is tracked with a particle-in-cell technique.

- 437 (CONF-761001--P3, pp 1027-1036) SAFETY ISSUES FOR ADVANCED-FUELED LMFBRs. Rose, D.; Deltrich, L.W.; Groimes, M.A.; Ferguson, D.R. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The safety issues for LMFBRs fueled with the candidate advanced-fuels, particularly for commercial size reactors, are reviewed and comparisons drawn with oxide-fueled fast reactors. The extent to which the definition and resolution of safety issues for advanced-fueled reactors draws on the current level of understanding for oxide-fueled reactors is highlighted. Areas of technology in which work is required and underway to resolve the safety issues are also highlighted.

- 438 (CONF-761001--P3, pp 1057-1066) CENTRAL COMPUTERIZED DATA BASE FOR LIQUID METAL FAST BREEDER REACTOR SAFETY CODES. Alter, H. (Energy Research and Development Administration, Washington, DC); Flanagan, G.F.; Greene, N.M. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The majority of safety computer codes used in the Liquid Metal Fast Breeder Reactor (LMFBR) safety studies have been developed with funds provided by the Energy Research and Development Administration (ERDA). Currently, the data used by these codes are collected by each user who must transcribe it for use in his study. This procedure, unfortunately, involves redundant effort and many times requires the user to "evaluate" physical property data, even though he has no expertise in this area. The Safety Analysis Computerized Reactor Data (SACRD) effort has been initiated to alleviate many of the difficulties cited above.

- 439 (CONF-761001--P3, pp 1069-1080) INFLUENCE OF FUEL-COOLANT-INTERACTIONS ON LOSS OF FLOW TYPE CORE DISRUPTIVE ACCIDENTS FOR THE SNR=300. Essig, C.; Froehlich, R.; Schmuck, P. (Kernforschungszentrum, Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The influence of fuel-coolant-interactions (FCIs) on loss of flow type core disruptive accidents has been studied for the fresh and irradiated (end of third cycle) SNR=300 Mark 1A cores. For the fresh core two accident simulations (without and with axial expansion reactivity feedback) have been considered. For

these simulations it has been shown that FCIs must only be considered for the disassembly phase of the accident and these FCIs do not lead to dangerous implosive effects. The highest energy releases are obtained if FCIs are neglected. For the irradiated core the accident has been simulated without axial expansion reactivity feedback. FCIs during the disassembly phase do not lead to dangerous implosive effects. Unfortunately, for an irradiated core, pin failures and FCIs cannot be excluded during the predisassembly phase of the accident. These phenomena are discussed and several reasons are given which lead to the conclusion that these failures and FCIs during the predisassembly phase have rather limited consequences for the irradiated SNR=300 core considered.

- 440 (CONF-761001--P3, pp 1081-1090) UNPROTECTED LOSS-OF-FLOW TRANSIENTS IN LARGE LMFBRs. Tzaros, C.P.; Barthold, W.P.; van Erp, J.B. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The dependence of the time to reach sodium boiling in an unprotected loss-of-flow transient in large LMFBRs on sodium density reactivity, radial thermal core expansion, and flow decay characteristics has been investigated. From the results obtained it appears to be possible to avoid sodium boiling completely if the following features, achievable within practical design limits, are incorporated (a) extended flow coastdown, (b) enhanced negative reactivity feedbacks due to radial thermal core expansion, (c) reduced positive reactivity feedbacks due to sodium density changes and the Doppler effect, (d) enhanced coolant flow due to natural circulation.

- 441 (CONF-761001--P3, pp 1091-1100) TREATMENT OF INCOHERENCIES AND UNCERTAINTIES IN LMFBR LOF ACCIDENT ANALYSIS WITH PARSEC. Burns, R.D. III; Ott, K.O. (Purdue Univ., Lafayette, IN). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

As basis for an investigation of the parameter and model uncertainties as well as of inter-subassembly incoherency effects, a computer program, PARSEC (Parametric Safety Evaluation Code) is being developed at Purdue University. The basic features of the first version, PARSEC-1, are described and some typical results are presented. Results of the investigations presented here show the importance of treating incoherency and uncertainty effects in analyzing severe LMFBR accidents. In contrast, the more benign accidents appear to be quite insensitive to incoherency effects. In addition, spectra of accident consequences are calculated by varying impact parameters within their range of uncertainty. This yields more detailed risk assessment information than single best-estimate and worst-case runs.

- 442 (CONF-761001--P3, pp 1101-1110) PIPE RUPTURE STUDIES FOR THE CRBR USING DEMO AND SAS. Hummel, H.H.; Kalimullah. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

It was found, in agreement with previous calculations, that the hot channel of the DEMO

code model of the CRBR is well into the boiling range for a double-ended primary pipe rupture at the inlet nozzle, and boiling is just about attained for the peak channel. The situation improves rapidly as the break moves away from the inlet nozzle because inertial and static pressures maintain inlet plenum pressure long enough to permit stored pin heat to be removed. The most important hot channel factor in producing boiling conditions is the one applied to pin power, with factors affecting fuel-clad gap conductance and coolant flow contributing important but smaller effects. Calculations of maximum coolant temperatures with DEMO and with SAS using the same power and flow histories are in surprisingly close agreement considering the coarseness of the mesh in the DEMO pin heat transfer calculation.

- 443 (CONF-761001--P3, pp 1111-1118) ANALYSIS OF POSTULATED FFTF PIPE RUPTURES. Markowski, F.J.; McCall, T.B.; Strawbridge, L.E. (Westinghouse Electric Corp., Madison, PA). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A detailed assessment of the FFTF Primary Heat Transport System (PHTS) piping has led to the conclusion that the integrity of the piping is assured such that there is no realistic potential for a rupture. Nevertheless, consistent with the practice of showing design margins even for hypothetical events, a spectrum of postulated PHTS ruptures has been analyzed. The analyses showed that upstream of the reactor vessel inlet downcomer, rupture areas of any size including a double-ended rupture could be tolerated with no core coolant boiling. At the most limiting location, the reactor inlet nozzle, rupture areas of 75 in.² and 55 in.² could be tolerated for three-loop and two-loop operation, respectively. This paper will present the following: (1) the criterion with which consequences of postulated pipe ruptures are compared; (2) the general transient response of the FFTF to postulated ruptures; and (3) the acceptable rupture sizes for the FFTF primary loop for a variety of conditions.

- 444 (CONF-761001--P3, pp 1121-1133) WORK-ENERGY CHARACTERIZATION FOR CORE-DISRUPTIVE ACCIDENTS. Marchaterre, J. (Argonne National Lab., IL); Marciniak, T.; Bratis, J.; Fauske, H.; Jackson, J. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The key result sought after in the analysis of core-disruptive accidents (CDAs) in fast reactors, is the potential damage that a given accident can impart to the system. A primary source of damage is the expansion of vapor from materials that have been severely overheated by a prompt-critical excursion. As the high temperature vapors expand, they can do work on the structures that comprise the primary system of the reactor. Thus, the terms "work potential", and "work-energy" are often used to characterize damage potential. Current practice in analyzing damage in fast reactor systems is to develop a pressure-volume relationship for the expanding materials and then explicitly calculate the work done on various structural components in the primary system using a coupled hydrodynamics/structural response code. Although this provides a rather direct analysis of the damage, various techniques have been used to calculate the required pressure volume relationships. Further, it is not always practical to carry

out such detailed analysis, and a variety of damage potential estimates have been, and continue to be used. This has often led to confusion as to what the actual damage potential for a given accident would be and how the various results reported in the literature are related to each other. The objective of this paper is to define the main techniques that have been used, and present a set of consistent calculations that compare the most important of these techniques for a range of reactor sizes and excursion severities.

- 445 (CONF-761001--P3, pp 1152-1161) CRITICAL CONFIGURATIONS AND SECONDARY EXCURSIONS FOR HIGH TEMPERATURE FUEL/STEEL MIXTURES. Pena, J.; Froehlich, R.; Schmuck, P. (Kernforschungszentrum, Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The amount of fuel necessary for obtaining neutronically critical configurations is studied. Critical spheres of fuel or of fuel/steel mixtures embedded in sodium are considered and the necessary amount of fuel is determined as function of Pu-enrichment, void fraction, and steel content. It is found that rather large fuel masses are necessary, especially for high void fractions (low smeared densities) or relatively high steel content. The calculations have been done by applying transport and diffusion theory by using the 26 group KFK-INR cross section set. Secondary excursions are analysed for high temperature fuel masses by using a two partner model, where one mass is resting on depleted UO₂ and the other mass is falling on top of the first one. High and low density systems are studied. Free fall, fall in sodium, and in addition a rather slow fall are considered. For high density masses large reactivity ramps, high pressure pulses, but very low thermal energy releases are encountered. For low density systems relatively low reactivity ramps, relatively low pressures, but high thermal energy releases are obtained. The calculations have been performed by using the neutronic-hydrodynamics program KADIS and by starting from rather pessimistic assumptions. For the SNR=300 it can be concluded that secondary excursions in the external core retention device can be contained from a thermal and very likely also from a mechanical point of view if reasonable assumptions are made.

- 446 (CONF-761001--P3, pp 1172-1182) ROLE OF WALL HEAT TRANSFER AND OTHER SYSTEM VARIABLES ON FUEL COMPACTION AND RECRITICALITY. Dhir, V.K.; Castle, J.N.; Catton, I.; Kastenbergh, W.E.; Doshi, J.B. (Univ. of California, Los Angeles). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The assessment of the molten fuel gaining recriticality after a hypothetical core disruptive accident in a fast reactor is an important safety consideration. Recriticality of the disrupted core can be envisioned to occur, if the fuel rearranges itself into a denser configuration either due to gravity slumping of the molten fuel or due to pressure or heat transfer driven compaction of the earlier dispersed fuel. In this paper the role played by wall heat transfer, internal radiation and the bottle pressure on the physical state of the molten fuel pool is discussed. It is suggested that in the absence of a solid crust the heat transfer process from the molten fuel to the surrounding steel will

be very efficient because of melting and buoyancy driven removal of less dense steel through the pool of heavier UO_2 . The internal radiation at the high fuel temperature significantly increase the effective thermal conductivity of the molten fuel and lead to increased heat transfer in situations where a solid crust of UO_2 exists between molten UO_2 and molten steel. IN a boiled-up bottled pool, the pool pressure is shown to increase very rapidly with time and thus necessitate higher fission heating of the fuel to maintain it in a certain boiled up state. Finally, the results of the above discussion are applied to study the recriticality of a fuel pool formed during a hypothetical core disrupted accident in a fast reactor.

- 447 (CONF-761001--P3, pp 1216-1224) LARGE-SCALE DETERMINISTIC COMPUTER CODE SYSTEMS AND LMFBR SAFETY: A PERSPECTIVE. Wider, H.U.; Bohl, W.R.; Dunn, F.E.; Ferguson, D.R. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The development and application of large-scale deterministic computer code systems in the LMFBR safety field involves a set of complex and many-faceted issues. This paper discusses a number of these issues. Motivations cited for developing these code systems are the desire to predict more realistic accident sequences and the need to isolate and identify key phenomena which require further experimental and analytical investigation. Code development difficulties which are discussed include those arising from the limited capabilities of current computer systems, the sometimes limited understanding of relevant phenomena and the need to provide some experimental verification of code models. The interaction required among code developers, experimentalists and phenomenological modelers is explored. In addition, the roles that such safety code systems can play in LMFBR design and in the actual licensing process are discussed.

- 448 (CONF-761001--P3, pp 1225-1235) SAS4A LMFBR ACCIDENT ANALYSIS CODE SYSTEM: A PROGRESS REPORT. Ferguson, D.R.; Bohl, W.R.; Bowers, C.H.; Cahalan, J.E.; Dunn, F.E.; Heames, T.J.; Kyser, J.M.; Wang, W.L.; Wider, H.U. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The SAS4A LMFBR accident analysis code system, which is currently under development at Argonne National Laboratory as the successor to the SAS3A code system, is discussed. Significant experience gained in the application of SAS3A to the study of hypothetical core disruptive accidents in a variety of LMFBR designs which has motivated the overall approach taken in SAS4A is cited. The structure and organization of SAS4A is presented, followed by a description of the new general code modules and phenomenological modules that are being developed for SAS4A.

- 449 (CONF-761001--P3, pp 1244-1252) MULTI-FIELD HYDRODYNAMIC DISASSEMBLY CALCULATIONS WITH SPACE-TIME KINETICS: FX2/VENUS-III. Weber, D.P.; Ganapol, B.D. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The development of computer codes which

analyze the hydrodynamics, neutronics, and thermodynamics of hypothetical LMFBR disassembly starting with the Lagrangian 2-D hydrodynamics code VENUS and extending to the Eulerian hydrodynamics code FX2/VENUS-III RM which includes treatment of multi-field motions as well as space-time kinetics are reviewed. The physical assumptions that form their basis are indicated and the differences in solution technique are illustrated. Primary emphasis is given to the description of the Eulerian code development with indications of the capabilities of implicit differencing, the necessity of tracking generic fuel types, the effect of relative motion, and the role of space-time kinetics. Several examples and comparisons of codes are given for FTR and 1000 MW(e) designs.

- 450 (CONF-761001--P3, pp 1253-1260) CORE DISRUPTIVE ACCIDENT AND RECRITICALITY ANALYSIS WITH FX2-POOL. Abramson, P.B. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The current state of development of FX2-POOL, a two-dimensional hydrodynamic, thermodynamic and neutronic scoping model for Hypothetical Core Disruptive Accident analysis is described. Checkout comparisons to VENUS for prompt burst conditions were good. Use of FX2-POOL to examine the importance of fuel to steel heat transfer during a prompt burst indicates that heat transfer plays no important role on that time scale. Scoping studies of material thermohydrodynamics for about 20 to 30 milliseconds following the prompt burst indicate that heat transfer is important on the time scale necessary for the CDA bubble to grow to the size of the original core. Preliminary results are presented for energetics of boiling fuel steel pools which are forced recritical by local surface pressurization.

- 451 (CONF-761001--P3, pp 1269-1279) PROBABILISTIC/DETERMINISTIC PROCEDURE FOR ANALYZING LMFBR CORE DISRUPTIVE ACCIDENTS. Vaurio, J.K.; Mueller, C. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A general procedure is presented for obtaining the probability distributions of selected consequences of an LMFBR hypothetical core disruptive accident. The uncertainties of the consequences are considered as a variability of the system and model input parameters used in the accident analysis. Probability distributions are assigned to these input parameters and parameter values are systematically chosen from these distributions. These input are then used in deterministic consequence analyses which are performed by fast-running analogues of the comprehensive mechanistic accident analysis codes. The results of these deterministic consequence analysis are used to generate the coefficients for multivariate polynomials which approximate the consequences in terms of the selected input parameters. These approximating polynomials are then used to generate the probability distributions of the consequences with random sampling being used to obtain the accident parameters from their distributions. The use of the procedure is illustrated by an application to a postulated loss of flow transient with failure to scram in the CRBR.

- 452 (CONF-761001--P3, pp 1285-1294)

GENERALIZED EULERIAN METHOD IN REACTOR CONTAINMENT ANALYSIS AND ITS COMPARISON WITH OTHER NUMERICAL METHODS. Chu, H.Y.; Wang, C.Y.; Gvildys, J.; Chang, Y.W. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The implicit continuous-fluid Eulerian method has been extended to include the treatment of the irregular cell which contains a moving wall boundary and the treatment of the special cell which has an out-flow boundary condition where the opening of the out-flow boundary is smaller than the side of the Eulerian cell. The extended method has been applied to a number of problems of interest in fast reactor containment analysis. Numerical results of sample calculations are presented. Comparisons with other numerical solutions and experimental measurements are also given.

- 453 (CONF-761001--P3, pp 1295-1303) EVALUATION OF EXPLICIT FINITE-DIFFERENCE TECHNIQUES FOR LMFBR SAFETY ANALYSIS. Bernstein, D.; Golden, R.D.; Gross, M.B.; Hofmann, R. (Science Applications, Inc., Oakland, CA). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

In the past few years, the use of explicit finite-difference (EFD) and finite-element computer programs for reactor safety calculations has steadily increased. One of the major areas of application has been for the analysis of hypothetical core disruptive accidents in liquid metal fast breeder reactors. Most of these EFD codes were derived to varying degrees from the same roots, but the codes are large and have progressed rapidly, so there may be substantial differences among them in spite of a common ancestry. When this fact is coupled with the complexity of HCDA calculations, it is not possible to assure that independent calculations of an HCDA will produce substantially the same results. Given the extreme importance of nuclear safety, it is essential to be sure that HCDA analyses are correct, and additional code validation is therefore desirable. A comparative evaluation of HCDA computational techniques is being performed under an ERDA-sponsored program called APRICOT (Analysis of Primary Containment Transients). The philosophy, calculations, and preliminary results from this program are described in this paper.

- 454 (CONF-761001--P3, pp 1314-1323) SIMULATION OF HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS IN VESSEL MODELS. Egleme, M. (Belgonucleaire, Brussels); Brahy, N.; Fabry, J.P.; Lamotte, H.; Stievenart, M.; Holtbecker, H.; Actis-Dato, P. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Explosion experiments have been performed with a pyrotechnic compound in overstrong vessels, without internals or with a perforated plate, and in yielding vessels. Thanks to these tests the charge characteristics have been determined, the pressure-flow characteristics of a perforated plate attached to the roof have been established and the validity of the SURBOUM-II code has been checked for various geometries of yielding vessels.

- 455 (CONF-761001--P3, pp 1324-1333) COMPARISON OF ANL CONTAINMENT CODES WITH SNR-300 SIMULATION EXPERIMENTS. Marchertas, A.H.;

Wang, C.Y.; Fistedis, S.H. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A comparison of REXCO and ICECO code predictions is made with data obtained from experiments of LMFBR excursion models. The comparisons are based on published results of tests conducted for the safety analysis of the SNR-300 fast breeder. The test configurations consist of a centrally located spherical source immersed in a pool of water which is encased in a cylindrical container. The cylindrical walls of the container are prestressed by hold-down bolts which span the two rigid ends. The space above the surface of the water within the container is occupied by air. Although certain aspects of the tests could not be simulated by the analytical models exactly, the comparison of results shows quite close agreement. The fact that the REXCO and ICECO codes involve different analytical formulations, their own close correspondence of results lends added credence to the value of analytical predictions.

- 456 (CONF-761001--P3, pp 1334-1343) SAFETY ANALYSIS OF THE PRIMARY HEAT TRANSPORT SYSTEMS OF LMFBRs. A-Moneim, M.T.; Chang, Y.W. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A two-dimensional Implicit Continuous-Fluid Eulerian Code designed for the safety analysis of the primary heat transport systems of LMFBRs is described. The code includes a separate model for all the components of the system, except the primary pump. These models are coupled together hydrodynamically to form a loop that can be subjected to two simultaneous pressure pulses. The structural response of the external walls of all the pipes and components is coupled with the hydrodynamics so that the fluid transients and the structural response of the system can be analyzed. Only the elbow walls are limited to be rigid.

- 457 (CONF-761001--P3, pp 1344-1353) SOME ASPECTS OF THE STRUCTURAL RESPONSE OF POOL-TYPE LMFBR SYSTEMS TO HCDA AND SEISMIC EVENTS. Marciniak, T.J.; Seidensticker, R.W.; Kulak, R.F.; Kennedy, J.M.; Fistedis, S.H. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Of the geometries possible in configuring an LMFBR system, there are essentially two which have been given the greatest amount of study throughout the world. These are the loop system and the pool system. This study addresses the safety related questions of the pool-type system to response energetic HCDAs and seismic events. Using representative reactor configurations of commercial size LMFBRs, it is shown that it is possible to design pool-type systems to accommodate HCDA and seismic loads despite the fact that the primary tank is a large diameter, thin walled structure and that the shield deck could be vulnerable to such loads.

- 458 (CONF-761001--P3, pp 1354-1362) MITIGATING EFFECT OF COOLANT CAVITATION IN THE RESPONSE OF LMFBR CONTAINMENT TO A WHOLE-CORE ACCIDENT. Jones, A.V. (Joint Research Centre, Ispra, Italy). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Oct 1976).

The deformation and damage to LMFBR containment consequent upon a whole-core accident is the result of the transmission of energy in the form of pressure waves from core region to containment structures by the coolant. If a mechanism can be invoked which dissipates some fraction of the energy of these pressure waves within the coolant itself the cost and other benefits in designing and constructing a containment for a given whole-core accident are obvious. This paper presents evidence that cavitation in the coolant is such a mechanism, recounts the theoretical developments made in modelling coolant behaviour and estimates the effect of cavitation in reducing loads to containment both in model experiments using water and in the real reactor. Plans for further theoretical and experimental investigations are also discussed.

- 459 (CONF-761001--P3, pp 1365-1375) FUEL-CLAD STRESS--STRAIN STATE AT THE BEGINNING OF A TRANSIENT. Sun, Y.H.; Ukrent, D. (Univ. of California, Los Angeles). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

This paper deals with KRASS (Kwick-Running Analysis for Stress and Strain), a simplified, fast-running, axis-symmetric computer code intended to predict fuel element conditions after long term steady state operation in an LMFBR. KRASS assumes restructuring has already occurred and divides the fuel pellet into two zones, an inner, highly plastic, hot region and an outer, cooler region which, together with the cladding, can undergo creep. The code allows for fission gas pressure, fuel swelling, and hot pressing with simple models, and it includes alternative correlations for stainless steel swelling. Fuel cracking is not included. The output of KRASS includes the radial and axial distribution of stress and strain on fuel and clad as a function of burnup. KRASS takes eleven hundred machine unit seconds on the UCLA IBM-360-91 to calculate fuel element behavior to 15 percent burnup with seven axial sections in the fuel column. The results indicate that the axial variation of fuel-clad mechanical interaction depends strongly on the fluence-to-burnup ratio, as well as the applicable stainless steel swelling correlation, and that transient fuel element behavior studies of pre-irradiated fuel need to allow for such differences. The position of maximum fuel-clad mechanical interaction frequently is found to occur in the lower third of the fuel element during steady state irradiation; at high fluence-to-burnup ratios the central axial nodes generally exhibit gap-reopening tendencies.

- 460 (CONF-761001--P3, pp 1376-1384) PREDICTION OF HIGH STRAIN-RATE DEFORMATION OF LMFBR CLADDING DURING REACTOR TRANSIENTS. Chow, L.S.H.; Shack, W.J. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A time-independent plasticity model is proposed. The model is incorporated into the LIFE code to predict the high strain-rate cladding deformation. A benchmark problem was solved using the modified code. The results are in good agreement with an analytical solution and a finite-element solution for the same problem. The modified code is also used to predict cladding deformation in simulated over-power transient experiments. In this case,

adequate agreement can only be obtained using a strain-rate-dependent yield stress.

- 461 (CONF-761001--P3, pp 1385-1392) RESPONSE OF SINGLE SUBASSEMBLY HEXCAN WRAPPERS TO PRESSURE LOADS. Ash, J.E. (Argonne National Lab., IL); Marciniak, T.J.; Romander, C.M.; Cagliostro, D.J. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The Liquid Metal Fast Breeder Reactor (LMFBR) core structure consists of a matrix of hexagonal subassembly ducts. Evaluation of the safety aspects of the core structure requires that reliable computational procedures be available to predict the deformation response of the subassembly configuration to postulated local energy releases. Finite-element computer codes have been developed to calculate deflections and strains of a hexcan subassembly wrapper subjected to internal and external dynamic pressure loadings. A series of experiments on simple out-of-pile hexcan sections has been performed to validate the computer codes. Particular attention has been directed towards the determination of the influence of the mechanical properties of the steel upon the hexcan response.

- 462 (CONF-761001--P3, pp 1399-1408) ANALYTIC STUDIES OF SUBASSEMBLY FAILURE PROPAGATION. Belytschko, T. (Univ. of Illinois, Chicago); Kennedy, J.M. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The prediction of subassembly response to energy sources and failure propagation is a three dimensional problem involving a fluid coupled with a structure. Because of the large expense involved in solving such three dimensional problems, a simplified, quasi-three dimensional treatment, in which the axial flow within the subassembly is treated by a one dimensional model coupled to a two dimensional model which represents the behavior in the cross section of the energy source has been developed in STRAW. For purposes of studying the validity and limitations of this model, a more complex computational approach which retains more of the three dimensional character was used and compared to STRAW predictions. It is shown that the results obtained by the quasi-three dimensional model are on the whole satisfactory; its predictions differ from the three dimensional solutions by about 10 percent or less in most cases.

- 463 (CONF-761001--P3, pp 1409-1418) STRESS INTENSITY FACTORS FOR CORNER-CRACKED SUBASSEMBLY DUCTS. Petroski, H.J. (Argonne National Lab., IL); Achenbach, J.D. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Plane strain stress intensity factors for cracks along the inside corners of internally pressurized hexagonal subassembly ducts have been calculated by various methods and compared.

- 464 (CONF-761001--P3, pp 1419-1426) EXPLOSION TESTS ON LMFBR-CORE MODELS; AN EVALUATION FOR REACTOR SAFETY. Will, H. (Institut fuer Reaktorentwicklung Gesellschaft fuer Kernforschung mbH, Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Oct 1976).

With regard to a certain, serious, unprobable reactor fault, explosion tests on core models for the SNR 300 are being carried out. The aim of the experiments is, to prove, that the deformation resulting from such a fault does not influence the reactor to be shut down. Until now, the six experiments that have been done show, that even in the case of conservative essentials with respect to the pressure loading of the postulated local fault, no power excursion can be expected. The deformations remain in the row of subassemblies next to the incident wrapper. The main experimental results are presented.

- 465 (CONF-761001--P3, pp 1427-1435) FINITE ELEMENT SOLUTION OF TRANSIENT FLUID-STRUCTURE PROBLEMS IN LAGRANGIAN COORDINATES. Donea, J.; Fasoli-Stella, P.; Giuliani, S. (EURATOM, Ispra, Italy). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A finite element method of analysis is described for the transient dynamic response of fluid-structure configurations. Hydrodynamic elements with linear and quadratic local velocity fields are considered. The structural elements are treated by a corotational formulation which accounts for large-displacement effects and nonlinear material behaviour. A lumped-explicit scheme is used for time integration of the equations of motion. Representative results are given for a Lagrangian frame of reference.

- 466 (CONF-761001--P4, pp 1587-1595) MODEL FOR PREDICTING TEMPERATURE FLUCTUATION CHARACTERISTICS BEYOND THE EXIT OF LMFBR SUBASSEMBLIES WITH FAULTS. Eifler, I.W.; Nijssing, I.R. (Commission of the European Communities, Ispra, Italy). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A model based on the theory of turbulent diffusion and including the effect of the coolant conductivity is described, which relates the main characteristics of the fluctuating temperature field in the plenum to those of the time average temperature profile at the bundle outlet, which is distorted as a consequence of a flow blockage. With the aim to show the potential of an early failure detection method based on temperature fluctuation measurements in the coolant plenum, the model has been applied to a number of practical situations. The applications concern the signal optimisation, "background" temperature fluctuations and signals caused by typical cooling anomalies.

- 467 (CONF-761001--P4, pp 2077-2086) EXPERIMENTAL INVESTIGATIONS ON THE COMPATIBILITY OF A SNR-TYPE CORIUM WITH GRAPHITE. Peehs, M. (Kraftwerk Union AG, Erlangen, Ger.); Friedrich, H.J.; Heuvel, H.J. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Tests on the compatibility between UO_2 -pellets, UO_2 /stainless steel mixtures (70 w/o UO_2 , 30 w/o SS) and graphite were carried out at 1600, 2000, and 2400°C. The test duration was 0.5 h. UO_2 -pellets are only completely converted to carbides at 2400°C, when liquid U-carbide phases occur. In the presence of SS the liquefaction temperature was lowered to 2000°C. The conversion rates in the case of UO_2 -

liquefaction as indicated by the corresponding CO-generation are high. The interactions are completed within a quarter of an hour. Apart from the release of O in the form of CO, the pick-up of C and the evaporation losses at the highest temperatures no significant changes in the chemical composition could be found. Post melting analyses by microsections, microprobe and X-ray investigations detected $UFeC_2$ -phases and carbides from the M_7C_3 -type together with austenite. A thermodynamic evaluation revealed that the C-activity of the melts after UO_2 -liquefaction is nearly 1.

- 468 (CONF-761001--P4, pp 1605-1614) LMFBR PROMPT BURST EXCURSION (PBE) EXPERIMENTS IN THE ANNULAR CORE PULSE REACTOR (ACPR). Schmidt, T.R. (Sandia Labs., Albuquerque, NM). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A series of in-pile experiments has been initiated to provide information on the energetics of LMFBR cores subjected to hypothetical superprompt-critical excursions. In these experiments, a single fuel pin in an instrumented piston-loaded autoclave is pulse heated into partial vaporization by the ACPR with initial reactor periods as low as 1.4 msec. Nine experiments have been performed with fresh unstructured UO_2 fuel pins. The first three were conducted without sodium coolant. In the dry experiments, energy depositions up to 3200 J/gm were obtained resulting in low pressure histories, negligible energy conversion ratios, and large upward axial fuel motions within the cladding. With sodium, the energy depositions also ranged up to 3200 J/gm resulting in large sharp pressure events (60 MPa), very low energy conversion ratios (2.1×10^{-4}), and within-cladding axial fuel motions comparable to the dry capsule experiments.

- 469 (CONF-761001--P4, pp 1615-1624) LOSS-OF-COOLING EXPERIMENTS WITH SINGLE FUEL PINS. Kwast, H. (Netherlands Energy Research Foundation ECN, Petten). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Some results are presented of three series of loss-of-cooling experiments, namely with (1) fresh fuel pins, (2) pre-pressurized fresh fuel pins, and (3) pre-irradiated fuel pins. These pins were tested in sodium filled capsules. Furthermore the Loss of Cooling (LOC) time and the type of the shroud tube (closed or with slots), were used as parameter. The results indicate that considerable clad melting can occur without leading to fuel failure. In cases fuel failure occurred this happened very gently. Preliminary results indicate that in case of equal energy input the extent of pin failure decreases by using a slotted shroud tube and pre-irradiated fuel pins. Considerable gas blanketing can occur if fuel pins with high internal pressure are subjected to a Loss of Cooling condition. The occurrence of a pressure pulse in case of an experiment with a low pressure, low burn-up fuel pin is most likely attributable to an FSI. So far no significant deformations of the capsule construction have been observed, so it can be concluded that the fuel failures were of a benign nature.

- 470 (CONF-761001--P4, pp 1625-1646) RESULTS OF RECENT TOP AND LOF EXPERIMENTS IN TREAT. Rothman, A.B.; Dickerman, C.E.; Lo, R.K.; Palm, R.G.; Semenza, L.A.; Simms, R.; Wider, H.U. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor

safety and related physics; Chicago, IL, USA (5 Oct 1976).

Tests E8, L6, F1 and F2 were recent experiments performed at TREAT as a part of the LMFBR Safety Test Program. Test E8 employed seven pins with fuel which had an axial variation of its fuel microstructure more like FTR than in previous tests. Test E8 was the first overpower experiment with clear indication of an initial postfailure fuel dispersal. An analysis made with the PLUTO code indicates that the driving force for fuel expulsion to the coolant channel was fission gas. The subsequent fuel dispersal could be explained by fission-gas expansion and a very mild fuel-coolant interaction.

- 471 (CONF-761001--P4, pp 1647-1657) SUMMARY AND EVALUATION OF R-SERIES LOSS-OF-FLOW SAFETY TESTS IN TREAT. Spencer, B.W.; Grohmes, M.A.; Holtz, R.E.; Testa, F.J.; Kramer, N.A. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A series of in-pile tests has been performed in which 7-pin bundles of FTR fuel pins (fresh UO_2) have been subjected to transient undercooling conditions representative of a hypothetical unprotected loss-of-flow (LOF) accident in FTR. The tests were undertaken to evaluate SAS modeling and code predictions of coolant voiding, molten cladding motion, and cladding blockage formations. Agreement with SAS predictions was found to be remarkably good. The absence of sodium superheat, sodium reentry, and fuel-sodium interactions was confirmed. Flooding of molten cladding by sodium vapor streaming was confirmed to result in an upper cladding blockage prior to subsequent formation of the lower blockage. The tests also revealed two areas where improvements in modeling were needed (1) to handle the additional effects upon coolant and cladding motion caused by plenum gas release, and (2) to improve the description of upward cladding motion and blockage formation which was found to be less substantial than predicted.

- 472 (CONF-761001--P4, pp 1658-1662) SIMULATION OF LOF ACCIDENTS WITH DIRECTLY ELECTRICAL HEATED UO_2 PINS. Alexas, A. (Kernforschungszentrum, Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The behavior of directly electrical heated UO_2 pins has been investigated under loss of coolant conditions. Two types of hypothetical accidents have been simulated, first, a LOF accident without power excursion (LOF accident) and second, a LOF accident with subsequent power excursion (LOF+TDP accident). A high-speed film shows the sequence of events for two characteristic experiments. In consequence of the high-speed film analysis as well as the metallographical evaluation statements are given in respect to the cladding meltdown process, the fuel melt fraction and the energy input from the beginning of a power transient to the beginning of the molten fuel ejections.

- 473 (CONF-761001--P4, pp 1679-1688) CLADDING FAILURE IN TREAT OVERPOWER EXPERIMENTS: A MECHANISTIC INTERPRETATION AND ITS IMPLICATIONS FOR LMFBR SAFETY ANALYSIS. Atcheson, D.B.; Sherry, R.R.; Shimane, K.J. (General Electric Co., Sunnyvale, Ca). 1976.

From International meeting on fast reactor

safety and related physics; Chicago, IL, USA (5 Oct 1976).

Eleven TREAT overpower experiments have been analyzed using the BEHAVE-SST fuel rod mechanics program. The results suggest that mechanistic prediction of cladding failure under hypothetical overpower conditions is feasible. Differential fuel-cladding thermal expansion and fuel expansion upon melting appear to be the dominant loading mechanisms in the experiments analyzed. It was necessary to postulate a fission gas loading mechanism in only one experiment to account for the observed behavior. These results suggest that failure noncoherence under overpower conditions will depend primarily on rod-to-rod variations in fuel strength, cladding strength and the distribution of available volume within the fuel.

- 474 (CONF-761001--P4, pp 1691-1696) CLADDING RELOCATION EXPERIMENTS. Henry, R.E.; Jeans, W.C.; Quinn, D.J.; Spieha, E.A. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

When analyzing the consequences of a hypothetical core meltdown accident in an LMFBR, the movement of molten clad material must be well understood for two reasons; (1) it has the potential for plugging the system as it leaves the core region, and (2) it is directly coupled to the neutronics of the system. To provide a detailed insight into this motion, small-scale single and ten pin tests have been conducted using Wood's metal to simulate the cladding material. In addition, a full length (FFTF) 28-pin triangular bundle experiment has been fabricated and is currently being calibrated. The objective of these experiments is to study the behavior of molten cladding within a voided subassembly where sodium vapor streaming tends to counteract the drainage due to gravity.

- 475 (CONF-761002--P4, pp 1697-1706) CLAD RELOCATION DYNAMICS: THE PHYSICS AND ACCIDENT EVOLUTION IMPLICATIONS. Theofarous, T.G.; Prather, W.; Chen, M.; Speis, T.P.; Lois, L. (Purdue Univ., Lafayette, IN). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Even under transient flow conditions the value $J^*/\text{sub } g/ = 1$ represents the threshold for upward film relocation. However, the two-phase frictional pressure drop multiplier is found to be substantially smaller than that predicted by the classical steady-state correlations for annular flow. Based on the values determined here new flow regime maps for clad relocation in LMFBR bundles under axial and radial melting incoherencies are presented. For large values of $J^*/\text{sub } g/$ upward relocation is usually accompanied by separation from the advancing film front and entrainment. The entrained mass proceeds at much higher velocities. Based on these findings and the calculated (by SAS) core pressure drop substantial clad relocation would be expected under LDFA conditions.

- 476 (CONF-761001--P4, pp 1707-1714) FUEL DISPERSAL EXPERIMENTS WITH SIMULANT FLUIDS. Farahat, M.; Henry, R.E.; Santori, J. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Boiling pool stability experiments have been

conducted for open and "bottled up" systems to ascertain the dispersive characteristics of such systems. These studies were performed in a microwave oven with water as the boiling liquid. High speed movies demonstrate that open pools are dispersive at very low vapor production rates and that "bottled up" pools, as a result of upward heat sinks created by the pressurization, are also dispersive at vapor production rates equivalent to 2 percent of nominal reactor power.

- 477 (CONF-761001--P4, pp 1725-1734)
TRANSIENT UNDERCOOLING ANALYSIS OF THE DISPERSAL MECHANISM TEST. Mync, L.; Bowers, C.H.; Heineman, J.B. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The initiating phase response of the proposed Dispersal Mechanism Test to an unprotected loss-of-flow initiator is studied with the SAS3A accident analysis code. This is done in order to study the question of whether or not the initiating phase would have a high probability of evolving into the gradual core meltdown/boilup transition phase scenario which this proposed test is intended to examine. It is found that the initiating phase could well terminate with a power burst driven by the slumping of fresh fuel. This has led to a search for core design changes which would tend to mitigate any such burst.

- 478 (CONF-761001--P4, pp 1735-1743)
EXPERIMENTS ON PRESSURE-DRIVEN FUEL COMPACTION WITH REACTOR MATERIALS. Henry, R.E.; Erickson, E.G.; Goldfuss, G.T.; Heiberger, J.J.; Quinn, D.J.; Roth, R.L.; Spieha, E.A.; Winsch, I.O. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Pressure driven, fuel-recompaction experiments have been conducted with reactor materials in 37-pin wire wrapped bundles. The experiments have been performed with the bundle full, half full, and empty of sodium coolant. Measurements and post-test examination of the bundles show that the sodium coolant had no significant effect on the final disposition of the fuel. Consequently, no fuel recompaction forces were observed.

- 479 (CONF-761001--P4, pp 1744-1753)
EQUATIONS OF STATE FOR LMFBR MATERIALS. Barnes, J.F.; DeVault, G.P. (Los Alamos Scientific Lab., NM). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Tabular equations of state are constructed for use by the SIMMER code in its investigations of LMFBR safety in hypothetical core disruptive accidents. A correction to a sodium equation of state reported earlier is described. The revised tabular data are presented. Need for further work on the melting transition is suggested.

- 480 (CONF-761001--P4, pp 1757-1766) STUDY OF FUEL FREEZING AND CHANNEL PLUGGING DURING HYPOTHETICAL OVERPOWER TRANSIENTS. Wong, K.W.; Dhir, V.K.; Kastenber, W.E. (Univ. of California, Los Angeles). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A mechanistic approach has been used to

determine the events following pin failure which may lead to fuel freezing and channel plugging. The location of the impact of the fuel particle with a wire wrap and the induced impact stresses in the particle are determined. A fuel particle is assumed to plate out at the impact location, if the induced impact stress exceeds the maximum fracture or yield stress. Knowing the distribution of the particles around the failure location, the amount and axial distance from the point of failure are determined. The effect on fuel plate-out of such variables as the rate and amount of fuel injection, fuel particle size, fission heating rate and presence of fission gas has been studied parametrically. The above modelling is applied to determine the amount and location of fuel plate out in CRBR under hypothetical slow and fast transients.

- 481 (CONF-761001--P4, pp 1767-1777)
HYDRODYNAMIC MODEL FOR TREATING TRANSIENT FREEZING IN PIN BUNDLES. Hakim, S.J.; Henninger, R.J. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A one-dimensional treatment of incompressible two-component flow in pin bundles is presented. The model monitors the temporal evolution of the pin geometry due to ablation, crust formation, and growth at every axial position along the flow channel. Heat conduction and generation in the pin structure are included. The model is used to extract the dependence of the penetration distance of the fluid up to the freezing temperature on inlet pressure, inlet temperature, initial pin temperature, pin diameter, and the hydraulic diameter of the flow. Comparison with available experimental results is made. The model was used to calculate the penetration distance of molten core material in the axial blankets of a liquid metal fast breeder reactor during a hypothetical core disruptive accident.

- 482 (CONF-761001--P4, pp 1788-1798)
ANALYTICAL AND EXPERIMENTAL STUDIES OF TRANSIENT FUEL FREEZING. Epstein, M.; Henry, R.E.; Grohmes, M.A.; Fauske, H.K.; Goldfuss, G.T.; Quinn, D.J.; Roth, R.L. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The freezing of a molten ceramic fuel-metal mixture in a single steel tube and in a seven-pin bundle has been studied experimentally. The freezing data in the single-tube experiment compare well with a simple transient heat-conduction model. It was found that the extent of penetration of a molten fuel-metal mixture into a seven-pin rod bundle is considerably greater than that predicted by existing models based on bulk freezing concepts. A tentative explanation for the observed penetration behavior is that, while melting of the steel channel structure rapidly leads to a fuel freezing in a bulk manner, complete occlusion of the pin bundle channels is ultimately controlled by conduction within the melted steel.

- 483 (CONF-761001--P4, pp 1808-1818)
DYNAMICS OF SOLIDIFICATION OF FLOWING FLUIDS WITH APPLICATIONS TO LMFBR POST-ACCIDENT FUEL RELOCATION. Chun, M.H.; Gasser, R.D.; Kazimi, M.S.; Ginsberg, T.; Jones, O.C. Jr. (Brookhaven National Lab., Upton, NY). 1976.

From International meeting on fast reactor

safety and related physics; Chicago, IL, USA (5 Oct 1976).

The process of solidification of a flowing fluid has been studied both analytically and experimentally. A series of tests was performed, using paraffin wax and Wood's metal as flowing hot fluids, to verify the model as well as to investigate experimentally the effects of various parameters on the process of transient flowing and freezing through a vertical channel. Preliminary experimental results for plugging times and the total mass of molten fluid displaced through the channel before plugging are presented. Effects of important parameters on the transient freezing process have been examined experimentally. A preliminary evaluation of the model is made based on comparison with the experimental results.

- 484 (CONF-761001--P4, pp 1833-1842) MECHANISMS OF VAPOR EXPLOSIONS. Bankoff, S.G.; Jo, J.H.; Ganguli, A. (Northwestern Univ., Evanston, IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A recalculation is presented of the Board-Hall theory of thermal detonation waves, taking into account the fuel-coolant mass ratio in determining the velocity equilibration time constant. The required peak pressure is now approximately 10^5 bar, so that fast droplet breakup by Taylor instability cannot be visualized. Crust breakup times of approximately 10^{-3} to 10^{-4} sec are calculated for a fuel drop entering the sodium coolant due to bubble growth and collapse, followed by jet penetration of the crust. Coarse premixing of the fuel and coolant is therefore difficult to attain. Taken together, these calculations imply that the probability of an efficient fuel-sodium explosion is very small. 26 references.

- 485 (CONF-761001--P4, pp 1843-1851) LIQUID--LIQUID CONTACT IN VAPOR EXPLOSIONS. Ochiai, M.; Bankoff, S.G. (Northwestern Univ., Evanston, IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A self-mixing theory for initiation and early propagation of vapor explosions is presented. Random contacts are postulated to cause local regions of high pressure at the contact interface, which result in annular jets tending to mix the fluids. The criterion for successful initiation depends upon the minimum Weber number for coalescence being exceeded over an area at least equal to the original contact area. Experimental free-fall droplet data are presented from which the minimum Weber number correlation is determined.

- 486 (CONF-761001--P4, pp 1852-1861) SOME ASPECTS OF MIXING IN LARGE-MASS, ENERGETIC FUEL-COOLANT INTERACTIONS. Cho, D.H.; Fauske, H.K.; Grolmes, M.A. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Certain aspects of mixing involved in large-mass fuel-coolant interactions are examined, considering the energy required as well as the dynamics of mixing. It is observed that: (1) an energetic, coherent interaction involving large masses of molten fuel appears extremely unlikely, unless the molten fuel is coarsely premixed with liquid coolant prior to the interaction and the interaction propagates

rapidly throughout the entire system, (2) the premixing that is likely to produce an energetic interaction is one in which molten fuel is initially separated from liquid coolant by a vapor layer, and (3) while details of the fine-scale intermixing leading to an energetic interaction are not fully understood, it would appear to be a one-step process. The observations along with consideration of accident situations seem to suggest that the possibility of a large-mass, energetic interaction is extremely remote in oxide-fueled LMFBR systems.

- 487 (CONF-761001--P4, pp 1862-1869) VAPOR EXPLOSION EXPERIMENTS WITH SIMULANT FLUIDS. Henry, R.E.; Fauske, H.K.; McUmber, L.M. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Several simulant fluid vapor explosion experiments have been performed to test the predictions of the interface temperature, spontaneous nucleation model. All experiments conducted to date have been in agreement with the model.

- 488 (CONF-761001--P4, pp 1870-1878) HEAT TRANSFER AND PRESSURE GENERATION DUE TO A TEMPERATURE DISCONTINUITY AT A LIQUID--LIQUID INTERFACE. Cooper, F.M. (Los Alamos Scientific Lab., NM). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

In the absence of a phase transition, the solution is presented for the time evolution of the temperature and pressure fields for fuel and coolant in both planar and spherical geometries assuming an initial temperature discontinuity. The problem is solved analytically using Laplace transforms. In spherical geometry the temperature of a sphere in a medium relaxes to the temperature of the medium in a time approximately R^2/α where R is the radius and α the thermal diffusivity of the sphere. For a sphere of fuel in coolant or coolant in fuel, with the fuel freezing, the time-dependent temperature distributions are determined and the motion of the freeze front is solved by using a boundary layer approximation.

- 489 (CONF-761001--P4, pp 1879-1888) A MODEL FOR FUEL FRAGMENTATION DURING MOLTEN FUEL/COOLANT THERMAL INTERACTIONS. Vaughan, G.J. (Culham Lab., Abingdon, Eng.); Caldarola, L.; Todreas, N.E. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Various physical models have been proposed to explain fuel/coolant interactions (FCI) and many sets of experimental data have been published. Some of the significant results are presented for a restricted class of FCIs. The experiments discussed are restricted to those in which one substance is dropped into the other as this mode of contact is the most quiescent. The model which is considered is a cyclic theory of bubble growth followed by collapse. The collapse, being asymmetric, forms a jet of coolant which can penetrate the fuel and vaporize forming another bubble and disrupting the fuel surface. 46 references.

- 490 (CONF-761001--P4, pp 1889-1895) THEORETICAL AND EXPERIMENTAL INVESTIGATION OF FUEL-COOLANT INTERACTIONS IN A SHOCK TUBE

CONFIGURATION. Goldammer, H.; Kottowski, H.M. (Commission of the European Communities, Ispra, Italy). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Experiments in a shock tube configuration were performed using molten stainless steel and UO_2 in combination with water, respectively sodium. The following data were measured as a function of time: (a) reaction force acting on the interaction crucible; (b) pressure of the vapor in the interaction space; (c) growth of vapor volume; and (d) ejection velocity. The experiments demonstrated that the process of "vapor explosion" in constrained configurations is dominated by successive ejection and reentry cycles. A physical model and a corresponding calculation code has been developed in order to simulate FCI in shock tubes.

- 491 (CONF-761001--P4, pp 1896-1904) EFFECTS OF FUEL-SODIUM INTERACTION IN TOP TESTS WITH ELECTRICALLY HEATED FUEL PINS. Bojarsky, E.; Deckers, H.; Drexler, H.; Gross, S.; Lehning, H.; Piel, D.; Reiser, H. (Kernforschungszentrum, Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

To simulate TOP accidents experiments are performed in an out-of-pile sodium loop using electrically operated UO_2 fuel pins. A single pin or a seven-pin bundle is heated from steady-state operation in a power transient of 300 to 400 ms duration so that up to 50 g of molten UO_2 penetrate into the cooling channel. Immediately after the pin has become defective narrow pressure pulses occur with amplitudes up to about 70 bar. They seem to be produced by collapsing of sodium vapor bubbles. Gas overpressures of about 2 to 3 bar in the fuel pin yield the highest amplitudes. Another type of pressure profile was observed, above all at a very low gas pressure in the fuel pin. It might be caused by evaporation of sodium penetrating into the defective zone. There is relatively small mechanical work done at the coolant during these events. A strong blockage of the cooling channel was observed within the damaged zone.

- 492 (CONF-761001--P4, pp 1907-1914) STATE OF THE ART IN AEROSOL MODELLING FOR LMFBR SAFETY ANALYSIS. Schikowski, W.O. (Kernforschungszentrum, Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The problems of aerosol formation, aerosol transport behavior, aerosol retention systems and (eventually) release to the environment from core disassembly accidents and other sources are of high significance in LMFBR safety. Beginning with the early models developed in the mid 60's much work has been done since to improve understanding of aerosol behavior in LMFBR containments. Present modelling theories comprise essentially the US code HARM-2 and the German code PARADISEKO-3. The status of aerosol modelling, the verification by experimental programs and the remaining uncertainties are discussed.

- 493 (CONF-761001--P4, pp 1915-1923) DEPOSITION OF AEROSOLS FORMED BY HCDA DUE TO DECAV HEAT TRANSPORT IN LMFBR INNER CONTAINMENT ATMOSPHERES. van de Vate, J.F.; Plomp, A.; de

Jong, C.; Smit, H.C.D. (Netherlands Energy Research Foundation ECN, Petten). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The role of heat transfer related processes governing aerosol behavior after core melt down in LMFBR containments is discussed. Useful and experimentally supported equations for calculation of aerosol decay due to thermophoresis or condensation coupled settling are given. The important aspect of spatially homogeneous distribution of aerosol in the containment space is treated. Comments are made about existing computer models for safety evaluation of nuclear aerosols.

- 494 (CONF-761001--P4, pp 1924-1929) FUEL PARTICLE AND FISSION PRODUCT RELEASE FROM LMFBR-CORE CATCHER. Jordan, S.; Ozawa, Y. (Kernforschungszentrum, Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

During a postulated severe incident with core rupture at a sodium cooled fast reactor major quantities of fuel particles may be suspended in the liquid sodium contained in the core catcher. The release of fuel and sodium from a contaminated pool at 500 and 900°C was investigated by laboratory and technical experiments. Sodium vapor was released from a boiling sodium pool into nitrogen atmosphere with a rate of 85 kg Na/m².h. Decontamination factors for uranium release from sodium pools were measured between 10³ and 10⁴ for sodium containing 1 weight percent UO_2 powder. The effect was found to be not strong dependent on temperature between 500 and 900°C.

- 495 (CONF-761001--P4, pp 1937-1944) SOME POTENTIAL REDUCTIONS IN THE RELEASE OF RADIOACTIVITY UNDER LMFBR ACCIDENT CONDITIONS. Nelson, G.T.; Johnson, R.P.; Vaughan, E.U.; Guderjahn, C.A.; Morewitz, H.A. (Atomic International, Canoga Park, CA). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The reactor system and containment of a LMFBR can be expected to be very effective in preventing the spread of radioactivity following an accident. With current leak data measurement technology, a reactor containment building must be considered to have 0.1 v/o leakage/day. Since this is the limit of sensitivity, this much gas leakage can be accommodated by flow through a single, short length hole having an equivalent diameter of about 1 mm. Recent tests at Atomic International have shown that holes this small in diameter are quickly plugged if aerosols are present to any extent. Since, in addition, radioactive volatiles will be mostly trapped in the sodium pool, there is little release of dose producing radioactivity.

- 496 (CONF-761001--P4, pp 1945-1954) SODIUM FIRE STUDIES IN FRANCE SAFETY TESTS AND APPLICATIONS ON AN LMFBR. Fruchard, Y.; Colome, J.; Malet, J.C.; Berlin, M.; de Cuy, G.D.; Justin, J.; Duco, J.; Fourest, B. (Centre d'Etudes Nucleaires de Cadarache, France). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The risk of sodium fires in an LMFBR requires thorough analysis, and the possible consequences of an accidental fire must be

accurately determined. Not only must means of prevention and detection be perfected, but techniques must be developed to limit the damage caused by a fire: extinguishment, aerosol containment, protection of reactor structures. The program currently undertaken by the SEA's Nuclear Safety Department covering these problems is described. The major results obtained as well as their application to the SUPER-PHENIX reactor are included.

from the primary system due to the system boundary failure, and (3) a characterization of events within the primary system that may affect the spatial distribution of radionuclides. The model is under development and has been applied to selected reactor designs and accidents. These applications provide insight into the reliability of the primary system to retain core debris, the range of accident outcomes, and the accident risks.

- 497 (CONF-761001--P4, pp 1955-1961) RELEASE OF FISSION PRODUCTS FROM CONTAMINATED SODIUM FIRMS. Jordan, S. (Kernforschungszentrum, Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Leaks in the primary coolant system of a LMFBR and also serious incidents with tank rupture may entail the escape of fission products into the containment of the reactor. For incident analysis it is important to know the retention capability of sodium for the different fission products. The release of cesium and strontium from pools contaminated with 10 to 1000 ppm was investigated by experiments. The cesium content of airborne aerosols depends on oxygen concentration: at 2 percent oxygen concentration the Cs content of sodium-oxide aerosols is 3 times and at 6.5 percent 15 times as high as the initial Cs concentration in the pool. Strontium content of aerosols over burning contaminated sodium pools is 10^3 times smaller than the strontium pool concentration.

- 498 (CONF-761001--P4, pp 1972-1981) SODIUM SPRAY RELEASE ACCIDENT ANALYSIS FOR FAST REACTOR SAFETY STUDIES. Shire, P.R. (Westinghouse Hanford Co., Richland, WA). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A computer code has been developed to model the effects of postulated sodium spray release from LMFBR piping, although this is an event of extremely low probability. The calculation method utilizes gas convection and droplet combustion theory to calculate the pressure and temperature effects in a range of 0 to 21 mole percent oxygen with humidity. Droplet motion and large aggregate sodium surface area result in rapid release of combustion and sensible heat causing a nearly adiabatic pressure rise which peaks in several seconds. This analytical tool has indicated reasonable agreement with prototypic test data for a range of oxygen and water vapor concentrations, cell volumes and droplet sizes.

- 499 (CONF-761001--P4, pp 1985-1994) MODEL FOR EVALUATING PROBABILITIES OF PRIMARY SYSTEM RETENTION OF CORE DEBRIS AND TIME OF RELEASE IN LMFBR RISK ASSESSMENTS. Fleischer, L.S.; Feller, K.G.; El-Sheikh, K.A.; Medley, J.R. (General Electric Co., Sunnyvale, CA). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

An analytical model is presented that can probabilistically analyze the capability of the primary coolant system to retain core debris after a core disruption accident. For a given set of conditions at the termination of a core disruption, the model provides (1) the probability of a failure of the reactor vessel and the primary system piping resulting from non-coolable core debris, (2) the probability distribution of the time of major fuel releases

- 500 (CONF-761001--P4, pp 1995-2004) CONSIDERATIONS OF THE THIRD LINE OF ASSURANCE: POST-ACCIDENT HEAT REMOVAL AND CORE RETENTION IN CONTAINMENT. Glueckler, E.L.; Dayan, A. (General Electric Co., Sunnyvale, CA). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

LMFBR safety through defense in depth is provided by four lines of assurance (LOA 1 to 4). Primary emphasis in design is on LOA-1 or prevention of accidents, but the LMFBR can also be shown to possess inherent safety characteristics in the second, third and fourth lines. These lines would limit core damage, retain radioactive materials in the primary system cells and mitigate the consequences of very low probability events. The main concern of this paper is with LOA3. Functional requirements and pertinent considerations of core retention and decay heat removal are given. The scenario of a postulated accident is described from the stage of core debris dispersion to the final stage of debris settling in either the reactor vessel or at the bottom of the reactor cell. Parametric studies for conservative scenarios which assume liner failure and subsequent chemical reactions have been performed for early-sized LMFBRs. The containment response appears to be highly dependent on the structural integrity of the reactor cell liner and concrete dehydration rates in the case the liner has failed. Radiological assessments indicate that 10CFR100 guidelines can be met for all considered cases.

- 501 (CONF-761001--P4, pp 2026-2035) PREDICTION OF DRYOUT HEAT FLUXES IN BEDS OF VOLUMETRICALLY HEATED PARTICLES. Dhir, V.K.; Catton, I. (Univ. of California, Los Angeles). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

New data for dryout heat fluxes in volumetrically heated shallow beds are reported. The shallow bed data are correlated with a semi-theoretical correlation based on maximum possible vapor volume flux through the vapor jets overlying the bed. The transition from deep to shallow beds is determined by using the present shallow bed correlation and the deep bed correlation suggested earlier. It is found that the transition height is significantly dependent on the particle size, and the initial bed porosity. The results of the present study are applied to a hypothetical core disruptive accident in a liquid metal fast breeder reactor. The maximum possible coolable depths of the core debris as a function of the particle size, bed porosity and decay heat have been predicted.

- 502 (CONF-761001--P4, pp 2036-2044) STUDIES OF THE FORMATION AND COOLING OF PARTICULATE FUEL DEBRIS BEDS IN SODIUM. Sowa, E.S.; Gabor, J.D.; Baker, L. Jr.; Pavlik, J.R.; Cassulo, J.C.; Holloway, W. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor

safety and related physics; Chicago, IL, USA (5 Oct 1976).

Particulate debris-bed formation and heat transfer experiments have been conducted to provide information needed for postaccident heat removal analysis of sodium-cooled reactors. The particulate debris bed formation experiment, designated M-5, employed molten UO_2 generated by a thermite-type chemical reaction. The molten UO_2 flowed into a dry steel container that simulated the cavity below a reactor. Sodium flow into the cavity followed the fuel flow by about 0.2 s. Examination of the residue indicated that the molten UO_2 was completely fragmented by interaction with sodium to form a particulate bed. Particle distributions down to the submicron size range are reported. Bed heat transfer experiments were performed with metallic particulates in water. Three experimental techniques were used with the same particulate material, including (1) induction heating of the particulate, (2) Joule heating of the aqueous phase (made conducting by the addition of salts), and (3) bottom heating.

- 503 (CONF-761001--P4, pp 2047-2055) EXPERIMENTAL INVESTIGATIONS OF HEAT TRANSFER CHARACTERISTICS IN LIQUID LAYERS WITH INTERNAL HEAT SOURCES. Fieg, G. (Kernforschungszentrum, Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Stationary natural convective heat transport from internally heated horizontal fluid layers is studied. As compared to previous measurements the spectrum of boundary conditions has been extended to cover the range from adiabatic lower to adiabatic upper wall, including the special case of equal temperatures at the upper and lower wall. In addition the heat transport from an internally heated fluid layer which is superposed by a second, not internally heated and nonmiscible fluid layer, has been studied. Moreover, heat transfer measurements have been done in internally heated liquid layers through which gas bubbles are rising from the bottom.

- 504 (CONF-761001--P4, pp 2056-2065) HEAT REMOVAL FROM MOLTEN FUEL POOLS. Baker, L. Jr.; Chasanov, M.G.; Gabor, J.D.; Gunther, W.H.; Cassulo, J.C.; Bingle, J.D.; Mansoori, G.A. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Two experiments were conducted to study the distribution of internally generated heat in nonboiling pools of rectangular cross-section. In one experiment, the heated fluid was an aqueous solution, and in the other the fluid was molten uranium dioxide. Heat was generated by the passage of AC current through the pools. Analysis of the heat flux data from the water experiments led to an empirical model of pool heat transfer. The model takes account of the interactions of the vertical heat fluxes with the horizontal heat fluxes. The empirical model, when applied to the conditions of the molten UO_2 experiments, predicted significantly smaller downward heat fluxes than were observed. The increased downward heat fluxes in the molten UO_2 could possibly be caused by effects of partial transparency within the molten UO_2 .

- 505 (CONF-761001--P4, pp 2066-2076) HEAT TRANSFER AND FLUID DYNAMICS IN A VOLUME-HEATED

BOILING POOL. Gustavson, W.R.; Kazimi, M.S.; Chen, J.C. (Brookhaven National Lab., Upton, NY). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

An analytical and experimental investigation of the horizontal heat transfer from an internally heated boiling pool to a vertical wall has been conducted. A model is presented for the heat transfer based on combined modes of natural and forced convection flow in the liquid boundary layer at the vertical wall. The experimental results, using Joule heating of a $ZnSO_4$ - H_2O solution, indicate that the pool may develop a low-density 'foam-like' regime as well as a dense 'turbulent' regime. The analytic model compares favorably to the results in the 'dense' regime which show that the highest heat flux will develop near the upper edge of the pool. Application of the model to a boiling pool of UO_2 with a low level of boiling power (decay-heat level) suggests that the sideward heat transfer rates will limit the thickness of any fuel crust that may develop between the pool and the surrounding steel structure to a few millimeters. The stability and strength of such a crust have not been investigated.

- 506 (CONF-761001--P4, pp 2105-2114) INTERACTIONS OF LMFBR CORE DEBRIS WITH CONCRETE. Baker, L. Jr.; Cheung, F.B.; Chasanov, M.G.; Sowa, E.S.; Bingle, J.D.; Staahl, G.; Pavlik, J.R.; Holloway, W. (Argonne National Lab., IL). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Experiments were conducted to study each of the four processes of importance in the analyses of the interactions of LMFBR core debris with concrete, including: (1) concrete dehydration, (2) interaction with molten stainless steel, (3) interaction with molten UO_2 , and (4) interaction with liquid sodium. The dehydration experiments, performed with one type of limestone concrete, indicated that as a result of dehydration and moisture migration within the concrete, the effective thermal diffusivity at higher temperatures is several times that at room temperature. Based on the observed rate of thermal diffusion in concrete, a preliminary model of heat transfer in concrete has been developed.

- 507 (CONF-761001--P4, pp 2115-2121) ACOUSTIC DIAGNOSTIC TECHNIQUES FOR POST-ACCIDENT HEAT REMOVAL EXPERIMENTS. Sutherland, H.J.; Carlson, G.A.; Kent, L.A. (Sandia Labs., Albuquerque, NM). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Two acoustic techniques which have applications to reactor safety studies are described here. Both techniques are adaptations of the pulse-echo ultrasonic method. The first technique uses acoustic waves to measure temperature. It will be used to monitor the temperature of UO_2 from 2000°K to above the UO_2 melting point (3130°K). The second technique uses acoustic waves to measure erosion rates and its use is illustrated by monitoring the erosion of a concrete interface by a plasma jet.

- 508 (CONF-761001--P4) FAST REACTOR SAFETY AND RELATED PHYSICS. VOLUME IV. PHENOMENOLOGY. (American Nuclear Society, Chicago, IL; European Nuclear Society, Petit-

- Lancy (Switzerland)). 1976. Contract W-31-109-ENG-38. 705p. Dep. NTIS, PC A99/MF A01. From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).
Separate abstracts are included for 58 papers concerning single-phase flow and sodium boiling; sodium boiling and subassembly flow blockages; transient-overpower and loss-of-flow experiments; fuel and cladding behavior and relocation; fuel and cladding freezing; molten-fuel-coolant interaction; aerosols and fission product release, and post-accident heat removal. Thirteen papers have been previously abstracted and included in ERA.
- 509 (CONF-761001--P4, pp 1439-1447) MULTICHANNEL ANALYSIS FOR PREDICTING TEMPERATURES AND BOILING INITIATION IN LMFBR SUBASSEMBLIES DURING A FLOW TRANSIENT. Nijssing, R.; Eifler, W. (EURATOM, Ispra, Italy). 1976.
From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).
The transient thermo-hydraulic subchannel behaviour of LMFBR subassemblies subjected to loss-of-coolant-flow conditions, is analyzed. The temperature behaviour in the stage prior to boiling inception is described. The model applied is based on the description of a significant number of subchannels and associated fuel rods and takes account of transverse mixing between adjacent subchannels. An analytical approach is used for solving the energy equations pertaining to both the coolant and the solid material region. Illustrative results are given for various rod bundle configuration subjected to a prescribed flow decay.
- 510 (CONF-761001--P4, pp 1457-1466) INTEGRAL BOILING PHENOMENA UNDER RADIAL TEMPERATURE GRADIENTS IN A TWELVE ROD BUNDLE OF LMFBR GEOMETRY. Boswinkel, H.H.; Maarleveld, J.R.C.; Turkcan, E. (Netherlands Energy Research Foundation ECN, Petten). 1976.
From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).
Steady boiling patterns obtained from a partly heated twelve rod bundle are discussed. A comparison with existing boiling models gives preference to the two-phase flow patterns which are usually applied in calculations for water cooled reactors. An attempt is made to apply the results obtained to the core of a fast breeder reactor under loss of flow conditions.
- 511 (CONF-761001--P4, pp 1467-1476) NUMERICAL SIMULATION OF THE TWO-DIMENSIONAL BOILING (VOIDING) IN LMFBR SUBASSEMBLIES. Miao, C.; Theofanous, T.G. (Purdue Univ., Lafayette, IN). 1976.
From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).
Preliminary estimates of the two-dimensional voiding effects indicate that they are not likely to be very significant in affecting the time to flow reversal.
- 512 (CONF-761001--P4, pp 1477-1487) AN INVESTIGATION OF FORCED CONVECTION BOILING OF SODIUM AT LOW HEAT FLUXES IN ELECTRICALLY HEATED SEVEN PIN BUNDLES. Eisenhower, S.W.; Albrecht, R.W. (Univ. of Washington, Seattle). 1976.
From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).
Sodium boiling at low flow and heat fluxes can be expected in a gamma-heated, in-reactor experiment. The results of experiments under these conditions (10-40 W/cm², 10-50 gm/s) in electrically heated seven pin rod bundles are described. The type of boiling is seen to depend upon the amount of superheat, the degree of inlet throttling and several system dependent parameters. The possible forms of acoustic and neutronic signals from an in-core experiment are discussed in terms of the underlying thermalhydraulics and the experimental findings.
- 513 (CONF-761001--P4, pp 1488-1493) TRANSIENT SODIUM BOILING CHARACTERISTICS AT PUMP RUN DOWN. Kottowski, H.M.; Savatteri, C.; Mol, M. (Commission of the European Communities, Ispra, Italy). 1976.
From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).
The transient pattern of boiling in simple channel geometry for loss of coolant conditions was investigated at various heat flux levels and mass flow rates. Heat flux densities of up to 900 W/cm were chosen for the individual experiments and were kept constant during the whole run-down until dry-out was reached. The rate of change of flow during the run-down was determined by the pump control circuit.
- 514 (CONF-761001--P4, pp 1494-1501) STEADY STATE LIQUID METAL BOILING PRESSURE DROP CHARACTERISTICS. Kottowski, H.M. (Commission of the European Communities, Ispra, Italy); Mol, M.; Savatteri, C.; Costa, J. 1976.
From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).
Steady state Na-boiling experiments were performed in order to measure the boiling characteristics (pressure drop). These experiments were executed at various heat fluxes and mass flow rates up to dry-out. For the analysis of the experiments a steady state boiling computation code (ESSD) was developed. In a first approach a parametric study has been made using the various two-phase flow-models and friction multipliers proposed by Lockhart-Martinelli, Chen, Kalish and Peppier. A validation of the code with the experimental results was made.
- 515 (CONF-761001--P4, pp 1502-1510) INFLUENCE OF THE AXIAL DISTRIBUTION OF HEAT FLUX ON SODIUM BOILING IN FORCED CONVECTION. Meneghello, S.; Pezzilli, M.; Scarano, G. (Comitato Nazionale Energia Nucleare, Casaccia, Italy). 1976.
From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).
The axial distribution of heat flux and single phase transient influence on incipient boiling superheat and sodium boiling dynamics have been studied by two test series in forced convection. These experiments were carried out in the same conditions of velocity, pressure, inlet temperature and mean heat flux on two wire wrapped single pin test-sections, geometrically similar, having respectively flat and sinusoidal ($\phi/\text{sub max}/\phi/\text{sub min} = 1.7$) axial distribution of heat flux. Moreover the influence of the bundle geometry on sodium boiling is carried out on the basis of the preliminary experiments on sodium boiling in seven pins bundle.

- 516 (CONF-761001--P4, pp 1523-1531) SODIUM BOILING EXPERIMENTS IN A 19 PIN BUNDLE: TWO PHASE COOLANT DYNAMICS. Costa, J.; Menant, B. (Commissariat a l'Energie Atomique, Turis). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Permanent sodium boiling experiments have been performed in a 19 wire-wrapped pin bundle on the forced convection loop CFNa II of the Service des Transferts Thermiques at the Grenoble Nuclear Research Center. These experiments have shown that subcooled boiling, highly sensitive to the multidimensional aspects of bundles, is very noticeable: the singular coolant dynamics during subcooled boiling could lead to consequences such as burn-out and pin failure. Nevertheless, the behaviour of the sub-assembly during saturated boiling becomes grossly analogous to the behaviour of a monodimensional channel, and the pressure drops which develop in the adiabatic parts of the subassembly may be of primordial influence on subsequent events.

- 517 (CONF-761001--P4, pp 1532-1540) TEMPERATURE DISTRIBUTION AND LOCAL BOILING BEHIND A CENTRAL BLOCKAGE IN A SIMULATED FBR SUBASSEMBLY. Brook, A.J.; Huber, F.; Peppler, W. (Gesellschaft fuer Kernforschung mbH., Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A series of experiments has been carried out to investigate the effects of localised disturbance to the normal coolant flow in a fast reactor fuel element. The tests involved an electrically heated bundle of 169 pins, with a centrally located blockage extending over 49% of the flow area. Test section geometry corresponded to the SNR 300 Mk Ia fuel element. Measured temperature distributions behind the blockage agreed well with those measured in corresponding water experiments. The observed features of local boiling are discussed, and it is shown that a continued capability for cooling the blockage region is preserved, even with intensive local boiling.

- 518 (CONF-761001--P4, pp 1541-1549) THERMAL-HYDRAULIC CORRELATIONS OF A SIX-CHANNEL BLOCKAGE IN A SODIUM-COOLED SIMULATED LMFBR FUEL ASSEMBLY. Han, J.T.; Fontana, M.H.; Wantland, J.L.; Gnadt, P.A.; Smith, C.M. (Oak Ridge National Lab., TN). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The temperature measurements in the wake behind a six-channel internal blockage in a simulated CRBR fuel assembly have been studied and analyzed. The coolant residence time in the wake and the mass exchange rate between the wake and the adjacent free stream are calculated using a simple physical model. Correlations are obtained for the average coolant temperature rise in the wake, the maximum coolant temperature rise in the wake, the maximum (inner) cladding temperature and the mean Nusselt number in the wake with Reynolds numbers in the range of 10^4 less than or equal to Re less than or equal to 10^5 and the power per pin in the range of 16 to 33 kW/m. The results can be easily applied to predict the temperature rises in the wake behind a six-channel blockage located anywhere in the fuel assemblies of the CRBR and the FFTF, providing the blockage is located two rows or more away from the duct wall and has a negligible heat

generation rate. Furthermore, it should be pointed out that no sodium boiling nor pin failure occurred in the tests and therefore, the reactor can still be operated safely under this kind of blockage condition.

- 519 (CONF-761001--P4, pp 1560-1567) RECENT RESULTS OF A LOCAL BLOCKAGE EXPERIMENT IN A SODIUM COOLED ELECTRICALLY HEATED BUNDLE. de Vries, J.F. (Netherlands Energy Research Foundation ECN, Petten); Hoebe, J.C.; Dorr, B. 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

The test section represents a 60-degree section of a SNR-300 fuel element having a 70% flat type central blockage. Experiments have been performed under single-phase and local boiling conditions. A parameter study of the wake temperatures has been carried out. It has been found that the normalized vortex temperatures depend on the Reynolds number. Apart from the temperature maximum in the vortex another relative temperature maximum has been found downstream the vortex. The boiling experiments showed that stable single-bubble boiling was dominating and the cooling capacity in the vortex enough to avoid dryout in this region.

- 520 (CONF-761001--P4, pp 1568-1577) LOCAL BOILING OF SODIUM IN DOWNSTREAM OF LOCAL FLOW BLOCKAGE IN A SIMULATED LMFBR FUEL SUBASSEMBLY. Kikuchi, Y.; Daigo, Y.; Ohtsubo, A. (Power Reactor and Nuclear Fuel Development Corp., Ibaraki, Japan). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

Local sodium boiling in the downstream of a six-subchannel blockage in an electrically heated LMFBR fuel subassembly mockup is described. The first series of experiments were conducted to measure temperature distributions in the downstream of the blockage under non-boiling conditions. The measured temperature rise due to the blockage agreed fairly well with the calculation by the LOCK code. When the experimental results are extrapolated to reactor conditions, the temperature rise due to the six-subchannel blockage is estimated to be less than 250°C. The second series of experiments were performed to investigate local boiling phenomena. In the local boiling region, no flow instability was observed since the subchannels near the wrapper wall were filled with subcooled liquid. In the nearly bulk boiling region, however, considerable upstream voiding occurred and then the inlet flow was reduced, leading to final dryout.

- 521 (CONF-761001--P4, pp 1578-1586) DECAY HEAT REMOVAL FROM A PIN BUNDLE. Kaiser, A.; Peppler, W.; Straka, M. (Gesellschaft fuer Kernforschung mbH., Karlsruhe, Ger.). 1976.

From International meeting on fast reactor safety and related physics; Chicago, IL, USA (5 Oct 1976).

A number of sodium experiments have been carried out using an electrically heated seven pin bundle to investigate steady-state heat dissipation capability of a LMFBR fuel element under complete inlet blockage conditions. Test have been performed in a 37 pin bundle to assess the amount of heat which can be removed under natural convection conditions in a system consisting of tests section, by-pass and cold leg. The results of several test runs are reported. The results of the complete inlet blockage tests are compared with those of a

simple theoretical model. Because of the influence of the honeycomb grids, the capability for steady state heat dissipation is markedly reduced. Amounts of heat which can be transferred by boiling under natural convection conditions correspond broadly to reactor decay heat levels.

- 522 (CONF-761107--27) HEAT TRANSFER IN TUBE BUNDLES OF HEAT EXCHANGERS WITH FLOW BAFFLES INDUCED FORCED MIXING. AbuRomia, M.M.; Chu, A.W.; Cho, S.M. (Foster Wheeler Energy Corp., Livingston, NJ (USA)). 1976. Contract EY-76-C-15-2395; EY-76-C-03-0972. 22p. Dep. NTIS, PC A02/MF A01.

From Annual meeting of the American Society of Mechanical Engineers; New York, NY, USA (29 Nov 1976).

Thermal analysis of shell-and-tube heat exchangers is being investigated through geometric modeling of the unit configuration in addition to considering the heat transfer processes taking place within the tube bundle. The governing equations that characterize the heat transfer from the shell side fluid to the tube side fluid across the heat transfer tubewalls are indicated. The equations account for the heat transfer due to molecular conduction, turbulent thermal diffusion, and forced fluid mixing among various shell side fluid channels. The analysis, though general in principle, is being applied to the Clinch River Breeder Reactor Plant-Intermediate Heat Exchanger, which utilizes flow baffles appropriately designed for induced forced fluid mixing in the tube bundle. The results of the analysis are presented in terms of the fluid and tube wall temperature distributions of a non-baffled and baffled tube bundle geometry. The former case yields axial flow in the main bundle region while the latter is associated with axial/cross flow in the bundle. The radial components of the axial/cross flow yield the necessary fluid mixing that results in reducing the thermal unbalance among the heat transfer to the allowable limits. The effect of flow maldistribution, present on the tube or shell sides of the heat exchangers, in altering the temperature field of tube bundles is also noted.

- 523 (CONF-761107--28) ECONOMICAL REMOVAL OF SODIUM WATER REACTION PRODUCTS (SWRP) FROM AN LMFBR INTERMEDIATE HEAT TRANSPORT SYSTEM (IHTS). Anand, R.K.; Reynolds, R.G. (American Society of Mechanical Engineers, New York). 1976. 6p. American Society of Mechanical Engineers, New York.

From Annual meeting of the American Society of Mechanical Engineers; New York, NY, USA (29 Nov 1976).

The system study concludes that, after a sodium-water reaction in an LMFBR power plant, the hot sodium circulation process is an economical and practical method of removing the SWRP from the IHTS. It is proposed that the impure sodium produced by the hot sodium circulation process be purified by a special fast cleanup cold trap design which is shown herein. The hot sodium circulation process, using the special fast cleanup cold trap, is shown to save a significant amount of downtime during the life of each LMFBR plant. A development program for the recommended process is included.

- 524 (CONF-770401--29) SUMMARY OF THE ORNL SHIELD DESIGN SUPPORTING ANALYSIS FOR THE FFTF. Engle, W.W. Jr.; Mynatt, F.R.; Emmett, M.B. (Oak Ridge National Lab., TN (USA)). 1977. Contract W-7405-ENG-26. 9p. Dep. NTIS, PC

A02/MF A01.

From 5. international conference on reactor shielding; Knoxville, TN, USA (18 Apr 1977).

From 1969 through early 1976, Oak Ridge National Laboratory was requested by the Westinghouse Advanced Reactors Division (WARD) to analyze various shields WARD designed for the Fast Flux Test Facility (FFTF). To develop the proper methodology for such a large and complex system, it was necessary to improve the basic cross-section data sets for sodium and iron and to devise calculational techniques which would allow some regions to be treated in detail and would also allow the analyses to be carried out in steps that could be coupled to each other. These requirements necessitated the performance of two types of experiments: measurements of radiation transport through bulk samples (simple geometry) of important FFTF materials; and measurements of radiation transport through mock-ups of specific geometric regions of the FFTF. The successful analyses of these experiments allowed the application of the derived data sets and various combinations and adaptations of the DOT discrete ordinates and MORSE Monte Carlo transport codes to the FFTF system itself. The first series of calculations for the full system revealed that a cavity surrounding the reactor vessel provided a pathway for neutrons to stream up to the head compartment. Also, fissions in fuel elements stored in the outer region of the reactor vessel were found to contribute significantly to the cavity streaming. As a result, the major ORNL effort was expended on analyzing various designs of a concrete shield to be added in the upper region of the reactor cavity. With the inclusion of the shield and modifications to the reactor vessel support system, the dose rates above the closure head were reduced by a factor of approximately 10^4 .

- 525 (CONF-770545--2) DEVELOPMENT OF A 9 CR STEEL WITH IMPROVED STRENGTH AND TOUGHNESS. Bodine, G.C. Jr.; Chakravarti, B.; Harkness, S.D.; Owens, C.M.; Roberts, B.; Vandergriff, R.; Ward, C.T. (Combustion Engineering, Inc., Chattanooga, TN (USA)). 1977. Contract W-7405-ENG-26. 5p. Dep. NTIS, PC A02/MF A01.

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

A development program was initiated to provide an alternate code-approved advanced ferritic alloy for LMFBR use. Approximately 30 laboratory heats were made of the type 9Cr-1Mo-2Nb-0.05V-0.02W. In elevated-temperature properties the experimental alloys are superior to code-approved and candidate commercial steels. Improvements realized in weldability, toughness, and elevated-temperature strength and stability are discussed.

- 526 (CONF-770545--3) MATERIALS DATA REQUIREMENTS FOR THE HIGH-TEMPERATURE STRUCTURAL DESIGN OF 2 1/4 CR-1 MO STEEL COMPONENTS. Corum, J.M.; Dalcher, A.W. (Oak Ridge National Lab., TN (USA); General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). 1977. Contract W-7405-ENG-26. 5p. Dep. NTIS, PC A02/MF A01.

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

The materials data needs that can arise when a structural alloy is used for the first time in liquid-metal fast-breeder reactor system components are discussed from the standpoint of the high-temperature structural designer. Recent experience with 2 1/4 Cr-1 Mo steel is used as an example. A common structural

problem, involving repeated thermal transient loadings, is discussed, and the current high-temperature structural design methods and criteria used to design for such loadings are outlined. It is shown that the limiting design consideration for 2 1/4 Cr-1 Mo steel is likely to be the accumulation of creep-rupture damage during the relaxation of residual stresses arising from thermal transient loadings. It is concluded that although many data needs for a new alloy are readily apparent, others are unique to each particular material and are often identified only as the material is employed in actual design.

- 527 (CONF-770611--31) SENSITIVITY OF THE POWER DISTRIBUTION IN LARGE HETEROGENEOUS LMFBR DESIGNS. Tzanos, C.P.; Barthold, W.P. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 7p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; New York, NY, USA (12 Jun 1977).

In heterogeneous LMFBR designs consisting of consecutive core and blanket zones, the power distribution is very sensitive to enrichment distribution changes in the core zones. The purpose of the paper is to analyze this sensitivity in heterogeneous LMFBR designs of different degrees of coupling among the core zones.

- 528 (CONF-770611--34) NEUTRON AND GAMMA TRANSPORT EFFECTS BY HETEROGENEOUS CORE DESIGNS. Lam, S.K. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 6p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; New York, NY, USA (12 Jun 1977).

The use of diffusion theory for the prediction of power production near a reactor core-blanket interface and the assumption that gammas are absorbed in situ can lead to substantial errors. This is primarily due to the breakdown of Fick's law for neutron diffusion near the core-blanket boundary, and the gamma leakage from the core into the blanket. These considerations are more pronounced in a situation where a large number of internal blanket assemblies are present, such as in the large heterogeneous core designs. The power distribution is studied for both fission and gamma heating in a large heterogeneous LMFBR with 3 core zones separated by 2 internal blanket zones. Comparisons are made between diffusion and transport theory for neutron calculations, and between in-situ absorption and rigorous transport theory calculation for gamma heating.

- 529 (CONF-770611--35) POTENTIAL AND LIMITATION OF THE HETEROGENEOUS REACTOR CONCEPT. Barthold, W.P.; Beitel, J.; Khan, E.; Tzanos, C. (Argonne National Lab., IL (USA)). 1977. Contract W-7405-ENG-48. 7p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; New York, NY, USA (12 Jun 1977).

Characteristics of LMFBR type reactors with heterogeneous cores are described. Reactor kinetic and safety aspects of heterogeneous and homogeneous cores are compared.

- 530 (CONF-770611--36) SYSTEMATIC APPROACH FOR CONSTRUCTING LOW SODIUM VOID HETEROGENEOUS CORES. Tzanos, C.P.; Barthold, W.P. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 7p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting;

New York, NY, USA (12 Jun 1977).

In unprotected loss-of-flow transients in large LMFBRs sodium voiding may result in high ramp rates of positive reactivity addition leading to rates of energy release that may exceed the design limits of the containment. Therefore, there is an incentive in designing reactors of low sodium void reactivity to reduce the positive reactivity ramp rates arising from sodium voiding. A systematic way of constructing heterogeneous configurations that have a near zero value of sodium void reactivity is presented.

- 531 (CONF-770611--37) PIN DIAMETER OPTIMIZATION IN 1200 MWE HETEROGENEOUS VS. HOMOGENEOUS LMFBRs. Orzechwa, Y.; Turski, R.B.; King, M.J. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 6p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; New York, NY, USA (12 Jun 1977).

LMFBRs with internal blankets (heterogeneous reactors) are known for reducing the sodium void reactivity and increasing the breeding ratio. As for homogeneous reactors, the optimization of the fuel pin diameter for heterogeneous reactors is of great interest. The optimum pin diameter is obtained by changing the fuel pin diameter until the homogenized fuel volume fraction is the same as the optimum fuel volume fraction of the homogeneous core. The optimization of the fuel pin diameter with respect to doubling time for a loosely coupled 1200 MWe oxide heterogeneous reactor is described. The results are compared with those of a homogeneous reactor.

- 532 (CONF-770611--38) FLOW ORIFICING OF LARGE LMFBR CORES WITH INTERASSEMBLY HEAT TRANSFER. Khan, E.U.; Barthold, W.P. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-48. 7p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; New York, NY, USA (12 Jun 1977).

Core flow orificing is important because fuel lifetime (or burnup) is strongly dependent upon the peak cladding temperature. The peak cladding temperature in different assemblies can be made nearly equal by properly allocating the flow to each assembly through flow orificing. In the method described each orifice zone has the same peak cladding temperature at some time in life. This method is especially important for large heterogeneous LMFBR (parfait) cores where interassembly heat transfer effects become more important than for the corresponding homogeneous LMFBR. The method was applied to orifice the flow in both homogeneous and heterogeneous cores.

- 533 (CONF-770735--4) SELF-WELDING EVALUATION OF STELLITE 6 AND STELLITE 56 IN FLOWING SODIUM. Chang, J.Y.; Schrock, S.L.; Johnson, R.V. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1977. Contract EY-76-C-62-345-001. 31p. Dep. NTIS, PC A03/MF A01.

From International Metallographic Society annual meeting; Houston, TX, USA (19 Jul 1977).

The self-welding behavior of two similar materials, Stellite 6 and Stellite 56, in sodium are discussed. The materials were tested at temperatures from 850 to 1400°F for time periods up to six-months while immersed in flowing sodium. Contact stresses ranged from 6000 psi to 16,000 psi on contact areas from 0.35 to 0.47 square inches. All separation tests to determine the extent of self-welding

were conducted in a tensile mode. The surface morphologies of the samples before and after each test were measured. At temperatures of 1150°F and above, almost all the Stellite 6 specimens indicated a significant tendency toward self-welding within a relatively short period of time (one week). Stellite 56 couples also developed a strong self-weld bond at 1160°F after six-month exposure.

time varying loading is becoming more and more important. However, notorious discrepancy has been observed between various analytical predications and experimental results for the buckling of shells. Various theories have been proposed to explain these discrepancies. Most of these theories are concerned with two aspects: initial imperfections and asymmetric responses. Both theories do narrow the gap between theoretical and experimental results; however, the remaining discrepancy is still not small. Other possible causes of this discrepancy have to be studied; among them, the boundary conditions. It has been pointed out that the slip at the boundary may have noticeable effect on the transient behavior of a plate. In this paper, the effect of various boundary conditions on the dynamic instability of axisymmetric shells is studied using the numerical discretization technique--convective finite element method.

- 534 (CONF-770807--1, pp vp) COMPOSITE CONTAINMENT FOR NUCLEAR POWER. 1977. From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

A new concept is described in which a steel liner is employed which carries much of the primary membrane loads. This type of structure is similar in some aspects to the previously described system: a mat, lined with a thin plate on its top surface, is similar to concrete containment; a cylinder and hemispherical dome, made up of steel plate and concrete, is about 2.5 feet thick (the minimum required for radiation shielding). The governing equations have been derived accounting for the fact that the stresses in the steel liner are calculated in accordance with the shell equations for hoop and meridional strains and temperature and the fact that the reinforcing bars respond as linear members. Because of composite behavior hoop and meridional membrane strain compatibility is maintained between the steel liner and the reinforcing bars.

- 537 (CONF-770807--35) SUBASSEMBLY BOWING EXPERIENCE IN EBR-II. Bottcher, J.H.; Hoffman, G.L. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 12p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In reactor bowing of subassemblies has caused some handling difficulties at EBR-II, the extent of the problem increasing with the subassembly stiffness. Analysis of four particular bowed subassemblies with respect to their core environment and structural design has indicated that residual bow is primarily the result of radiation-enhanced creep during irradiation.

- 535 (CONF-770807--27) SOME TRENDS IN CONSTITUTIVE EQUATION MODEL DEVELOPMENT FOR HIGH-TEMPERATURE BEHAVIOR OF FAST-REACTOR STRUCTURAL ALLOYS. Pugh, C.E.; Robinson, D.N. (Oak Ridge National Lab., TN (USA)). 1977. Contract W-7405-ENG-26. 25p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The paper addresses some important features of the inelastic behavior of 2 1/4Cr-1Mo steel and indicates a mathematical framework that is capable of representing these types of response. While the constitutive model discussed embraces capabilities beyond those of equations presently used in design analyses; their implementation into practicable analysis methods (such as finite-element programs) is more demanding. For example, in the case of slow time-dependent deformations, the equations governing accumulation of the inelastic strain components and the evolution of the tensorial state variable α are intimately coupled. A part of recommending any such model for use in design must be a quantitative assessment of the economic feasibility of implementation.

- 538 (CONF-770807--37) STRUCTURAL DYNAMICS IN LMFBR CONTAINMENT ANALYSIS: A BRIEF SURVEY OF COMPUTATIONAL METHODS AND CODES. Chang, Y.W.; Gvildys, J. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 19p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In recent years, the use of computer codes to study the response of primary containment of large, liquid-metal fast breeder reactors (LMFBR) under postulated accident conditions has been adopted by most fast reactor projects. Since the first introduction of REXCO-H containment code in 1969, a number of containment codes have evolved and been reported in the literature. The paper briefly summarizes the various numerical methods commonly used in containment analysis in computer programs. They are compared on the basis of truncation errors resulting in the numerical approximation, the method of integration, the resolution of the computed results, and the ease of programming in computer codes. The aim of the paper is to provide enough information to an analyst so that he can suitably define his choice of method, and hence his choice of programs.

- 536 (CONF-770807--34) DYNAMIC INSTABILITY ANALYSIS OF AXISYMMETRIC SHELLS BY FINITE ELEMENT METHOD WITH CONNECTED COORDINATES. Hsieh, B.J. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 12p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The instability of axisymmetric shells has been used in engineering fields as a safety device such as the rupture discs used in the LMFBR (Liquid Metal Fast Breeder Reactor) design to relieve the excessive pressure caused by the water and sodium reaction when there is a leak in the piping system. Hence, the analysis of the instability of shells under

- 539 (CONF-770807--39) PARAMETRIC INVESTIGATION OF FRACTURE OF EBR-II DUCTS. Chopra, P.S.; Moustakakis, B. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 23p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Results of preliminary static and dynamic finite element fracture mechanics analyses that

were conducted to analytically simulate the dynamic fracture behavior of EBR-II ducts are presented. The loads considered are those that may arise because of rapid release of fission gases from a failed fuel element inside a duct, obtained from some previous tests and a recent analytical model. In spite of the motivation for the present work, the analytical procedures described may have a wider general application in the fail-safe design of structures.

- 540 (CONF-770807--39) EFFECT OF HIGH EXTERNAL THERMAL LOADING ON LMFBR SUBASSEMBLY INTEGRITY. Smith, J.L.; Helenberg, H.W.; Kilsdonk, D.J.; Pedersen, D.R.; Pierce, R.D.; Wilson, R.E.; Ariman, T. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 17p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

One of the postulated accidents to be considered in the evaluation of the safety and integrity of a Liquid Metal Fast Breeder Reactor (LMFBR) is a complete meltdown (loss-of-cooling) of a subassembly without reactor shutdown. One barrier to further propagation is the ducts of the adjacent subassemblies. The meltdown of a subassembly can result in a pressure loading and/or a thermal loading of the adjacent subassembly ducts. There are four potential failure modes of the adjacent duct: (1) failure as a result of coolant boiling inside the adjacent duct followed by melting of the duct, (2) cracking of the duct as a result of thermal stress, (3) deformation of the duct as a result of thermal stress followed by coolant boiling and melt through, and (4) weakening of the duct as a result of high temperature allowing pressure loading to cause duct failure. The purpose of this paper is to discuss the experimental investigation of the thermal loading of the adjacent subassembly ducts as a result of application of fuel to one flat.

- 541 (CONF-770807--40) THERMAL AND STRESS ANALYSES OF MELTDOWN CUPS FOR LMFBR SAFETY EXPERIMENTS USING SLSF IN-REACTOR LOOPS. Blomquist, C.A.; Ariman, T.; Pierce, R.D.; Pedersen, D.R. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 21p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

A description of a meltdown cup to be used in the SLSF in-reactor experiments is presented. Thermal analyses have shown that the cup is capable of containing and cooling the postulated quantities of molten fuel and steel. The basic loadings for stress analyses were defined and failure modes were determined. It was shown that both the maximum bending stress and maximum tangential stress in the Inconel vessel are below the material yield stress. Additionally, the axial stress in the Inconel vessel was found to be negligible. The shear stress in the wire-formed retaining ring is much below the maximum shear stress. Therefore, the meltdown cup is capable of performing its required function.

- 542 (CONF-770807--41) DEVELOPMENT OF A FINITE-STRAIN FUEL-ELEMENT MODEL. Jankus, V.Z.; Shack, W.J. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 11p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The LIFE-III computer code was developed to calculate the thermal and mechanical response of mixed-oxide fuel elements in a fast-reactor environment. It incorporates a one-dimensional, steady-state heat-transfer analysis and a structural analysis based on generalized plane strain. Temperature, stress, and strain can be calculated for any specified history of reactor power cycling. Up to seven axial segments are permitted to account for axial variations in power and coolant temperature. Fuel restructuring, migration of fabricated porosity, fuel and cladding swelling and creep, fission-gas release, and fuel cracking, crack healing, and hot pressing are included in the analysis. The mechanical analysis in the original version of the code was based on the assumption that the strains in the fuel and cladding could be considered infinitesimal. However, this assumption is grossly violated in the vicinity of the central void of the fuel, since the opening and closing of the void involves displacements of the order of the void radius. To provide a more accurate description of the mechanical behavior of the fuel element, a finite-strain fuel-element model has been developed. The finite-strain calculations in the modified code are based on an incremental analysis. The deformation history of the fuel element is analyzed in discrete time steps, and the basic geometry of the element is updated after each time step. To perform the calculation for nonlinear materials, each axial segment is divided into a number (less than or equal to 20) of concentric cylindrical shells. The analysis maintains the overall force equilibrium of each ring and enforces geometric compatibility between the shells, and the boundary pressures and displacement increments at the inner and outer surfaces of a shell are related to the stresses and the strain increments within the shell in a novel manner.

- 543 (CONF-770807--43) MECHANICAL RESPONSE OF FFTF REFERENCE AND P1 CLADDING TUBES UNDER TRANSIENT HEATING. Youngahl, C.A.; Ariman, T.; Lepacek, B.E. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 13p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Burst tests of Type 316 stainless steel cladding tube samples subjected to increasing temperature and relatively constant internal pressure were conducted to assist in the pretest analysis of the P1 experiment performed in the Sodium Loop Safety Facility. This paper reports and analyzes the burst test results and those of subsequent transient heating work. The use of a modified extensometer in obtaining mechanical response data for stainless steel in the high temperature range is illustrated, some of such data is provided, and the potential of further experiments and analysis is indicated. Tubing of the same design as Fast Flux Test Facility (FFTF) cladding (20% cold worked, 0.230 in. OD, 15 mil wall) was tested as-received and after annealing or electrolytic thinning. P1 tubing (38% cold worked, 0.230 in. OD, 10 mil wall) was tested before and after aging under conditions anticipated in the P1 reactor experiment. The P1 cladding was designed to simulate FFTF tubing that had experienced irradiation embrittlement and attack by cesium oxide and sodium impurities.

- 544 (CONF-770807--44) PROOF TESTS OF IRRADIATED AND UNIRRADIATED EBR-II SUBASSEMBLY DUCTS. Ruther, W.E.; Chopra, P.S.; Lambert, J.D.B. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 15p. Dep.

NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

A series of dynamic pressure tests have been conducted within EBR-II subassembly ducts. The tests were designed to simulate bursting of a driver-fuel element in a cluster of such elements at their burnup limit during off-normal conditions in EBR-II. The major objective of the tests was to assure that such failure, which might cause rapid release of stored fission gas, would not deform or otherwise damage subassembly ducts in a way that would hinder movement of a control rod. The test results are described.

- 545 (CONF-770807--45) COMPUTER SIMULATION OF LMFBR PIPING SYSTEMS. A-Moneim, M.T.; Chang, Y.W.; Fistedis, S.H. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 10p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Integrity of piping systems is one of the main concerns of the safety issues of Liquid Metal Fast Breeder Reactors (LMFBR). Hypothetical core disruptive accidents (HCDA) and water-sodium interaction are two examples of sources of high pressure pulses that endanger the integrity of the heat transport piping systems of LMFBRs. Although plastic wall deformation attenuates pressure peaks so that only pressures slightly higher than the pipe yield pressure propagate along the system, the interaction of these pulses with the different components of the system, such as elbows, valves, heat exchangers, etc.; and with one another produce a complex system of pressure pulses that cause more plastic deformation and perhaps damage to components. A generalized piping component and a tee branching model are described. An optional tube bundle and interior rigid wall simulation model makes such a generalized component model suited for modelling of valves, reducers, expansions, and heat exchangers. The generalized component and the tee branching junction models are combined with the pipe-elbow loop model so that a more general piping system can be analyzed both hydrodynamically and structurally under the effect of simultaneous pressure pulses.

- 546 (CONF-770807--46) RESPONSE OF SUBASSEMBLY MODEL WITH INTERNALS. Kennedy, J.M.; Belytschko, T. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 23p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Analytical tools have been developed and validated by controlled sets of experiments to understand the response of an accident and/or single subassembly in an LMFBR reasonably well. They have been subjected to a variety of loadings and boundary environments. Some large subassembly cluster experiments have been performed, however little analytical work has accompanied them because of the lack of suitable analytical tools. Reported are analytical approaches to: (1) development of more sophisticated models for the subassembly internals, that is, the fuel pins and coolant; (2) development of models for representing three dimensional effects in subassemblies adjacent to the accident subassembly. These analytical developments will provide feasible capabilities for doing economical three-dimensional analysis not previously available.

- 547 (CONF-770807--47) STRUCTURAL RESPONSE OF LARGE LMFBR HEAD CLOSURES TO HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS. Kulak, R.F.; Marciniak, T.J.; Belytschko, T.B. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 18p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

A preliminary study of the dynamic response of a large (1200 MWe) pool-type deck structure to an energetic core disassembly accident was performed. The analysis proceeded as two independent, decoupled calculations: containment calculations and deck response calculations. The containment calculations were performed mainly to derive a loading pressure history for the deck structure. In part of the study, the deck was considered rigid so the derived loading pressures were conservative. In the second part of the study a set of independent deck response calculations were performed using the derived loading pressure history as input. A finite-element model that incorporated the main structural components of the deck was constructed. A preliminary study of the dynamic response of the deck was performed using this model. Also, the effect that different types of boundary conditions have on the dynamic response of the deck was studied. Based on the above preliminary studies, it is concluded that the deck structure can maintain its structural integrity under the given energy release.

- 548 (CONF-770807--49) EXTENSION OF HCDA SAFETY ANALYSIS TO LARGE PCRV CONTAINMENT STRUCTURES. Marchertas, A.H.; Fistedis, S.H.; Bazant, A.P.; Belytschko, T. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 29p. Dep. NTIS, PC A03/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The introduction of PCRV as part of LMFBR containment also brings in the need for dynamic analysis. The conventional design would not be adequate for designing PCRVs subject to highly transient loading. The PCRV models described provide a valuable illustration of the design arrived at by conventional means and loaded dynamically by means of the transient computer code. In particular the value of the computer code is realized in the case of the loop-type model. The original pressure-volume source used for the illustrative examples corresponds to a total available energy of 2720 MW-s, if the expansion were continued down to one bar. Thus the energy source is considerably larger than the energy releases utilized in the currently prevailing LMFBR safety evaluations. Although the resultant pressure acting on the walls of the PCRV varies with the particular design, the two sample cases indicate that no apparent difficulty is encountered in designing the PCRV to sustain those higher loads. The computer code is established as a valuable tool to study the PCRV response. It serves as a means to locate critical areas of the particular design. Parameter- or sensitivity studies could thus be made with great efficiency. As an example, one interesting and somewhat unexpected result of the illustrations presented is the occurrence of circumferential cracks through the walls of the PCRV.

- 549 (CONF-770807--50) FINITE-ELEMENT FORMULATIONS FOR THE THERMAL STRESS ANALYSIS OF TWO- AND THREE-DIMENSIONAL THIN REACTOR

STRUCTURES. Kulak, R.F.; Kennedy, J.M.; Belytschko, T.B.; Schoeberle, D.F. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 22p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In several postulated LMFBR subassembly-to-subassembly failure propagation events, it is hypothesized that the duct wall of an accident subassembly fails and deposits molten fuel on the outer wall of an adjacent subassembly. It is therefore necessary to determine if the deposited fuel will fail the adjacent wall and thus propagate the event. This entails a thermal stress analysis, and since at times the adjacent subassembly is internally pressurized, thermomechanical analysis are also of value. Solutions are presented for several elastic plastic thermal problems. Some of these examples are compared to available analytic solutions. In addition, the hypothetical accident of molten fuel deposition on the adjacent hexcan is addressed. Combinations of pressure and thermal loading are considered. It is shown that the principal feature of the response is a large in-plane compressive stress which would undoubtedly cause buckling.

550 (CONF-770807--51) EFFECT OF TUBE-SUPPORT INTERACTION ON THE DYNAMIC RESPONSES OF HEAT EXCHANGER TUBES. Shin, Y.S.; Jendzejczyk, J.A.; Wambagans, M.W. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 15p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Operating heat exchangers have experienced tube damages due to excessive flow-induced vibration. The relatively small inherent tube-to-baffle hole clearances associated with manufacturing tolerances in heat exchangers affect the tube vibrational characteristics. In attempting a theoretical analysis, questions arise as to the effects of tube-baffle impacting on dynamic responses. Experiments were performed to determine the effects of tube-baffle impacting in vertical/horizontal tube orientation, and in air/water medium on the vibrational characteristics (resonant frequencies, mode shapes, and damping) and displacement response amplitudes of a seven-span tube model. The tube and support conditions were prototypic, and overall length approximately one-third that of a straight tube segment of the steam generator designed for the CRBR. The test results were compared with the analytical results based on the multispan beam with 'knife-edge' supports.

551 (CONF-770807--52) SIMULATION OF LMFBR EXCURSION MODELS BY MEANS OF ICECO. Marchertas, A.H.; Wang, Y.C.; Fistedis, S.H. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 13p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Test data from experiments dealing with LMFBR containment studies are compared with containment code solutions. The features of fluid spillage, which is a subject of some concern in experiments, is modeled in the ICECO code and is stressed. Several variations of spillage solution were arrived at gaining more insight to this problem and thus serve in reducing some of the uncertainties connected with fluid spillage. Other features of the

analytical models are utilized and described.

552 (CONF-770807--53) ANALYSIS OF LMFBR CONTAINMENT RESPONSE TO AN HCDA USING A MULTIFIELD EULERIAN CODE. Chu, H.Y.; Chang, Y.W. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 13p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

During a hypothetical core disruptive accident (HCDA), a core meltdown may cause the fuel cladding to rupture and the fuel fragments to penetrate into the sodium coolant. The heat in the molten fuel may cause the liquid sodium to boil, changing its phase. The interactions between materials are so complicated that a single-material model with homogenized material properties is not adequate. In order to analyze the above phenomena more realistically, a Multifield Implicit Continuous-Fluid Eulerian containment code (MICE) is being developed at Argonne National Laboratory (ANL) to solve the multifield fluid-flow problems in which the interpenetrations of materials, heat transfer, and phase changes are considered in the analysis. The hydrodynamics of the MICE code is based upon the implicit multifield (IMF) method developed by Harlow and Amsden. A partial donor-cell formulation is used for the calculation of the convective fluxes to minimize the truncation errors, while the Newton-Raphson method is used for the numerical iterations. An implicit treatment of the mass convection together with the equation of state for each material enables the method to be applicable to both compressible and incompressible flows. A partial implicit treatment of the momentum-exchange functions allows the coupling drag forces between two material fields to range from very weak to those strong enough to tie the fields completely. The differential equations and exchange functions used in the MICE code, and the treatment of the fluid and structure interactions as well as the numerical procedure are described. Two sample calculations are given to illustrate the present capability of the MICE code.

553 (CONF-770807--54) EULERIAN ALGORITHM FOR ANALYZING FLUID-STRUCTURE INTERACTION IN THE FAST REACTOR CONTAINMENT. Wang, C.Y.; Chang, Y.W.; Fistedis, S.H. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 18p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The Eulerian algorithm described is capable of analyzing two-dimensionally the fluid-structure interaction in a typical LMFBR containment consisting of various structure components. The algorithm has several special features. First, it employs the Eulerian coordinates in the formulation and enables the technique to be applicable to excursions involving large material distortions. Second, it provides rigorous hydrodynamic analyses at the fluid-structure interface and at the geometrical discontinuities, which is very important in the study of wave-transmission and -diffraction problems. Third, the algorithm incorporates a well-developed finite-element structural program that can account for both material and geometrical nonlinearities. The application of this Eulerian algorithm is not limited to fast reactor containment. It can also be applied to the major piping components, such as valve and IHX, where internal baffle plates may play an important role in the wave propagation. Moreover, the treatment of the

time-dependent irregular cell can be integrated with the IMF² Method and structural dynamics program for investigating the coupled fluid-structure problems involving two-phase flows.

- 554 (CONF-770807--55) PTA-1 COMPUTER PROGRAM FOR TREATING PRESSURE TRANSIENTS IN HYDRAULIC NETWORKS INCLUDING THE EFFECT OF PIPE PLASTICITY. Youngdahl, C.K.; Kot, C.A. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 20p. Dep. NTIS, PC A02/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Pressure pulses in the intermediate sodium system of a liquid-metal-cooled fast breeder reactor, such as may originate from a sodium/water reaction in a steam generator, are propagated through the complex sodium piping network to system components such as the pump and intermediate heat exchanger. To assess the effects of such pulses on continued reliable operation of these components and to contribute to system designs which result in the mitigation of these effects, Pressure Transient Analysis (PTA) computer codes are being developed for accurately computing the transmission of pressure pulses through a complicated fluid transport system, consisting of piping, fittings and junctions, and components. PTA-1 provides an extension of the well-accepted and verified fluid hammer formulation for computing hydraulic transients in elastic or rigid piping systems to include plastic deformation effects. The accuracy of the modeling of pipe plasticity effects on transient propagation has been validated using results from two sets of Stanford Research Institute experiments. Validation of PTA-1 using the latter set of experiments is described briefly. The comparisons of PTA-1 computations with experiments show that (1) elastic-plastic deformation of LMFBR-type piping can have a significant qualitative and quantitative effect on pressure pulse propagation, even in simple systems; (2) classical fluid-hammer theory gives erroneous results when applied to situations where piping deforms plastically; and (3) the computational model incorporated in PTA-1 for predicting plastic deformation and its effect on transient propagation is accurate.

- 555 (CONF-770807--Pt.E, pp vp) ANALYSIS OF DYNAMICALLY LOADED NON-LINEAR STRUCTURES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

A CODES; CONCRETES; CONTAINMENT; LMFBR TYPE REACTORS; NONLINEAR PROBLEMS; REACTOR VESSELS; DYNAMIC LOADS

- 556 (CONF-770807--Pt.E, pp vp) ANALYSIS OF MECHANICAL HCDA CONSEQUENCES WITH ARES FOR DEMONSTRATION OF PRIMARY CONTAINMENT INTEGRITY IN SNR-300. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

A CODES; CONTAINMENT; FINITE DIFFERENCE METHOD; HYDRODYNAMICS; REACTOR CORE DISRUPTION; SNR-1 REACTOR; TIME DEPENDENCE

- 557 (CONF-770807--Pt.E, pp vp) ACOUSTIC WAVE MEASUREMENTS IN REACTOR-GRADE CONCRETES. 1977.

From 4. international conference on structural mechanics in reactor technology; San

Francisco, CA, USA (15 Aug 1977).

In this paper, five reactor grade concretes were investigated during their initial cure period to gather information which is a suitable data base for interpreting the results of confirmatory reactor safety research programs. In particular, the concretes studied were the concrete proposed for the Clinch River Breeder Reactor (CRBR), which is a limestone mix with flyash, and two other limestone-aggregate mixes and two basaltic-aggregate mixes. The curing process in these concretes was followed with strength tests and with acoustic-wave velocity measurements. The choice of the latter technique was based on the successful programs of several investigators who used acoustic waves to follow the curing process in concrete and this information was also required for instrumentation techniques that are proposed for accident simulation studies. The technique used to measure the sound velocities of concrete is a modified version of the pulse transmission technique introduced by Mattaboni and which conforms to ASTM Standard C597-71 for the measurement of the sound velocity in concrete. This technique uses an electronic technique to measure the transit time of an acoustic wave through a sample and, thus, its acoustic velocity. The results of the CRBR concrete program illustrate that the water retention stabilizes after 10 days of cure with a net loss of 0.4%; the addition of flyash to the mix stabilizes it and slows down the initial curing process from 33 to 90 days; the addition of the admixtures to the mix reduces its sound velocity from 5.74 km/s to 5.64 km/s and its strength from 51.6 MPa (7590 psi) to 42.1 MPa (6190 psi); and the sound velocity in this concrete stabilized after 75 days of cure.

- 558 (CONF-770807--Pt.E, pp vp) FINITE-ELEMENT FORMULATIONS FOR THE THERMAL STRESS ANALYSIS OF TWO- AND THREE-DIMENSIONAL THIN REACTOR STRUCTURES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper describes finite-element formulations for the thermal stress analysis of LMFBR structures. The first formulation is applicable to large displacement rotation problems in which the strains are small. For this formulation, a general temperature-dependent constitutive relationship is derived from a Gibbs potential function and a temperature dependent yield surface. The temperature dependency of the yield surface is based upon a temperature-dependent, material-hardening model. The model uses a temperature-equivalent stress-plastic strain diagram which is generated from isothermal uniaxial stress-strain data. A second formulation is presented for problems characterized by both large displacement-rotations and large strains. Here a set of large strain hypoelastic-plastic relationships are developed to linearly relate the rate of stress to the rate of deformation. The temperature field is described through time-dependent values at mesh node points; the temperature fields in each element are then obtained by interpolation formulas. Hence, problems with both spatial and temporal dependent temperature fields can easily be treated. The above developments were incorporated into two ANL developed finite-element computer codes: the implicit version of STRAW and the 3D Implicit Structural Analysis Code. STRAW is a two-dimensional code with a plane stress/plane strain beam element. The 3D Implicit code has a triangular flat plate element which is capable of sustaining both membrane and bending loads. To insure numerical

stability both codes are based on an iterative-incremental solution procedure with equilibrium checks based on an error in energy.

559 (CONF-770807--Pt.E, pp vp) COMPUTER SIMULATION OF LMFBR PIPING SYSTEMS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

A two-dimensional coupled hydrodynamic-structural response analysis of piping systems is described. Implicit Continuous-Fluid Eulerian (ICE) technique is utilized in the hydrodynamics while a finite-element technique is used in the structural analysis. Different piping components such as elbows, valves, reducers, expansions, heat exchangers, and tees are modelled and coupled with the straight pipe model. An axisymmetric general component model that can be used in modelling valves, reducers, expansions, and heat exchangers is described. At the inlet and outlet region of such component the cross-sectional area may change suddenly or gradually, or may not change at all. Among the options available in this model are deformable exterior walls, interior rigid wall simulation, and tube bundle effect. Exterior walls of pipes and components are treated as thin axisymmetric shell. A convected coordinate explicit finite-element scheme for large displacement small strain, elastic-plastic material behavior in which membrane and bending strengths are accounted for is employed. The strains are linearly related to the displacement of the element relative to its convective coordinates, and similarly, the nodal forces are linearly related to the elements stresses. The coupling of the hydrodynamics and structural problems is done in such a way that the hydrodynamics supplies the structure with a pressure loading and the structure supplying the hydrodynamics with a moving boundary condition. Because of the difficulties of handling interior walls that may occupy partial zones, the walls are assumed rigid and limited in their orientation to be parallel to the radial or axial directions, their position to zone boundaries, and their thickness to zero.

560 (CONF-770807--Pt.E, pp vp) MATERIAL PROBLEMS IN ACCIDENT ANALYSIS OF PRESTRESSED CONCRETE REACTOR VESSELS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

One problem investigated is the nonlinear constitutive behavior and failure criteria for concrete. Previously, a comprehensive theory, called endochronic theory, has been shown to satisfy all basic currently known features of test data. Nevertheless uncertainty still exists with regard to non-proportional loading paths, for which good test data are lacking at present. An extension of the endochronic theory which correlates best with general experimental evidence and includes fracturing terms is given, and a comparison with vertex-type hardening in plasticity is made. Finally, application in a finite element analysis of compression failure of a wall due to localized strain-softening bands is discussed. A second problem which must be analysed in accident situations is the high temperature shock on the concrete walls (due to liquid sodium, up to 850°C). Refining a previous crude formulation a rational model for calculating moisture and heat transfer and pore pressures in concrete subjected to thermal shock is presented. It employs full thermodynamic properties of water, considers dehydration of concrete, changes in porosity and in sorption isotherms, as well as

heat convection and thermal moisture flux. The model is calibrated by means of test data. For the purpose of failure analysis of the vessel, creep at high temperature, including the transitional thermal creep must be considered. Finally, a model in which the formulations for heat and moisture transport, for creep and for nonlinear stress-strain behavior are combined to calculate compression failure due to a thermal shock, is outlined. The ultimate purpose is the prediction of spalling, possibly the explosive spalling, of the wall.

561 (CONF-770807--Pt.E, pp vp) VALIDATION OF EXPLOSION CONTAINMENT CODES: ANALYSIS OF BARE RIGID MODELS IN THE COVA PROGRAMME. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

C CODES;CONTAINMENT;EXPLOSIONS;LMFBR TYPE REACTORS;REACTOR ACCIDENTS;MATHEMATICAL MODELS

562 (CONF-770807--Pt.E, pp vp) RESPONSE OF SIMPLE, SCALE MODEL REACTOR VESSELS TO A SIMULATED HCDA LOADING. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The work described here was performed to verify modeling techniques used in two-dimensional axisymmetric fluid-structure-interaction codes. A well-defined energy source was applied in simple reactor models having well-defined material properties, dimensions with precision tolerances, and simple boundary conditions, and the loading, strain, and deformations were measured. The experiments were performed in thick-walled and thin-walled vessels that are approximately 1/30-scale of the Clinch River Breeder Reactor (CRBR). The sodium coolant is simulated by water, and the HCDA load is simulated by the expansion of the detonation products of a low-density explosive detonated in the vessel core barrel. The detonation products expand from an initial pressure of 3700 psi down to 360 psi for a volume change of 410 cm³, equal to the cover gas volume. The expansion produces 2.87 kW-sec of gas work up to slug impact on the vessel cover. Piezoelectric quartz pressure transducers measure the loading pressure in the core and on the core support platform, the vessel wall, and the cover. Foil strain gages measure circumferential and axial strains on the vessel wall. The plastic work absorbed by the vessel before and after slug impact is calculated from the strain measurements and the post-test deformation profiles. In the thick-walled vessel, the peak pressures are 3000 psi on the core support platform, 6000 psi on the vessel wall, and 14,000 psi on the cover. The cover pressure rapidly drops to 7000 psi corresponding to slug impact pressure. In the thin-walled vessel, the platform and wall pressures are 50% lower and the cover pressure is only 7% lower.

563 (CONF-770807--Pt.E, pp vp) CHARGE DEVELOPMENT AND ANALYSIS OF RECENT UK EXPERIMENTS IN THE COVA MODEL TEST PROGRAMME. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The COVA programme of small scale experiments consists of a two pronged attack on the provision of high quality accurate data for validating explosion containment codes. This paper examines some effects arising from the inclusion in the model geometries of either

thin or over strong outer tanks, of hemispherical or right cylindrical bases or of simple internals. The main energy source for the COVA programme is a low pressure high explosive which has been especially developed by the UK for use in model test series. The paper describes the characterisation of this low density explosive which is based on a PETN/polystyrene mix expanded, under strictly controlled heating, to form self supporting charges of various sizes. This characterization, involving the analysis of cylinder tests and under-water firings of bare spheres, shows that the explosive performs in a reproducible manner in both dry and wet environments. The development of a JWL equation of state will be illustrated and discussed. The explosion containment codes being developed within the UK are the 2-D Lagrangian code ASTARTE-3 and the 2-D Eulerian code SEURBNUK-2, the latter being currently developed jointly with JRC, Ispra. Both codes are able to deal with transient compressible flow in deformable primary containment tanks, with various geometries of internal components, depending upon the limitations of Lagrangian or Eulerian concepts and the current state of the development of both codes. For the bare tank experiments the ASTARTE code can be used to predict the entire history of impulse loading due to the fluid flow and some typical comparisons are given for experiments with thin outer tanks.

564 (CONF-770807--Pt.E, pp vp) STRUCTURAL DYNAMICS IN LMFBR CONTAINMENT ANALYSIS. A BRIEF SURVEY OF COMPUTATIONAL METHODS AND CODES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper gives a brief survey of the computational methods and codes available for LMFBR containment analysis. The various numerical methods commonly used in the computer codes are compared. It provides the reactor engineers to up-to-date information on the development of structural dynamics in LMFBR containment analysis. It can also be used as a basis for the selection of the numerical method in the future code development. First, the commonly used finite-difference expressions in the Lagrangian codes will be compared. Sample calculations will be used as a basis for discussing and comparing the accuracy of the various finite-difference representations. The distortion of the meshes will also be compared; the techniques used for eliminating the numerical instabilities will be discussed and compared using examples. Next, the numerical methods used in the Eulerian formulation will be compared, first among themselves and then with the Lagrangian formulations. Special emphasis is placed on the effect of mass diffusion of the Eulerian calculation on the propagation of discontinuities. Implicit and explicit numerical integrations will be discussed and results obtained from these two techniques will be compared. Then, the finite-element methods are compared with the finite-difference methods. The advantages and disadvantages of the two methods will be discussed in detail, together with the versatility and ease of application of the method to containment analysis having complex geometries. It will also be shown that the finite-element equations for a constant-pressure fluid element is identical to the finite-difference equations using contour integrations. Finally, conclusions based on this study will be given.

565 (CONF-770807--Pt.E, pp vp) MODELING OF REACTOR INTERNALS IN THE HCDA ANALYSIS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper deals with the modeling of the reactor internals in the HCDA analysis. Internals which must be included in the analysis are core blankets, shields, fission-gas plenum, core barrel and core support structure. This is because they are placed so close to the reactor core that the expansion of the reactor core is strongly influenced by the presence of these internals. The general approach to the mathematical model is to incorporate, to the extent possible, the actual geometries and the correct material properties. In the HCDA analysis, the mechanism for producing mechanical damage to the primary containment is the expansion of the reactor core. Therefore, it is very important to model the core gas properly. The inertia of the core gas must be included. Four to six zones are needed for a proper representation of the core. The core blankets and shields can also be represented by zones. If they are segmented, they must be modeled as solid materials with no circumferential strength. Modeling of the core blankets and shields as hydrodynamic materials can lead to underestimation of the slug impact loading on the reactor head as well as the upper vessel wall deformation. Sliding lines must be provided at both sides of the core blankets and shields. The fission-gas plenum can be treated as a group of mixed materials containing gases, liquids and solids. Core barrels must be modeled as thin shells. If the core barrel is to be treated as continuum solid material, the thickness of the core barrel will have to be divided into at least three to five zones. The core support structure can be modeled either as a lattice of shell elements similar to the actual structure or as an elastic-plastic solid continuum material with overall bending stiffness equivalent to the actual structure.

566 (CONF-770807--Pt.E, pp vp) FAIL-SAFETY OF LMFBR STEAM GENERATOR SYSTEMS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

LIQUID METALS; LMFBR TYPE REACTORS; REACTOR ACCIDENTS; SODIUM; STEAM GENERATORS; WATER; FAILURES

567 (CONF-770807--Pt.E, pp vp) ANALYSIS OF LMFBR CONTAINMENT RESPONSE TO AN HCDA USING A MULTIFIELD EULERIAN CODE. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper describes a computer code, MICE (Multifield Implicit Continuous-fluid Eulerian Containment Code), which is being developed at Argonne National Laboratory (ANL) for the analysis of containment response to a hypothetical core disruptive accident (HCDA). The code is applicable to multifield flow problems where material fields are allowed to have penetrations. Reactor structures are treated as axisymmetrical shells and solved by the large-displacement and small-strain theory. The numerical technique used for the solution of the multifield fluid flow code consists of two principal steps. In the first step, the material fields in each cell are treated independently; coupling effects are evaluated by binary interactions between fields. In the second step, a pressure equilibration is performed among all fields in each cell; volume fractions are also adjusted in this step. A

partial donor-cell form is adopted for calculating the convective fluxes to minimize the truncation errors; a Newton-Raphson formula is used for the numerical iteration. Implicit treatment of the mass transport and state equations allows the flow speeds to range from far subsonic (incompressible flow) to supersonic. The coupling drag forces between fluids can vary from very weak to those strong enough to tie the fields together completely. Phase transition and heat transfer between fields are also considered into calculation. Two sample problems have been performed using the NICE code. The first illustrates the relative motions of the material fields after the initiation of a core disassembly accident. Core support structure and core barrel are modelled as rigid obstacles. The second demonstrates the interactions between fluid and structures.

- 568 (CONF-770807--Pt.E, pp vp) TRANSIENT RESPONSE OF FAST-REACTOR CORE SUBASSEMBLIES TO PRESSURE LOADS: A COMBINED ANALYTICAL AND EXPERIMENTAL APPROACH. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The intent of the paper is to describe a combined analytical-experimental programme that is being performed at J.R.C. Ispra in the field of subassembly deformation. From the numerical side, 2D and 3D non-linear dynamic finite element codes are being developed for coupled hydrodynamics structures situations. The structural part of these codes is based on the use of flexural elements in which non-linearities arising from elastic-plastic behavior and from large-displacement effects are accounted for. As far as the fluid flow phase is concerned, a Lagrangian code is used, since, due to the particular configuration of the subassemblies, the fluid distortions can be assumed to be relatively small. Nevertheless special hybrid elements are used on the interfaces between the liquid and the hexcan walls. In parallel with the analytical work experiments are performed in steps of increasing complexity to that each component of the computer codes can be adequately tested. The experimental series starts with single internally loaded hexcans in which the pressures, strains and displacements are measured as a function of time. The programme continues by explosively loading hexcans surrounded by one or two rows of water filled subassemblies which are held together by a rigid heavy ring. This configuration has well defined boundary conditions, which are necessary for code validation purposes. Experiments are also planned in which the fuel element pins and their spacers will be modelled. Comparisons between experimental and analytical results will be presented for a single internally loaded subassembly and possibly for a subassembly surrounded by a first row of water filled hexcans.

- 569 (CONF-770807--Pt.E, pp vp) ANALYSIS OF MECHANICAL HCDA CONSEQUENCES IN THE COOLANT LOOPS FOR DEMONSTRATION OF PRIMARY CONTAINMENT INTEGRITY IN SNR 300. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The demonstration of the accommodation of the mechanical consequences of a hypothetical core disruptive accident (HCDA) within the primary system is an essential objective in the safety assessment for SNR-300. For this purpose a mathematical model has been developed, in which the pressure pulse propagation in fluid-filled

piping systems is calculated. This model takes into account the compressibility of the fluid as well as the elastic-plastic deformation of the pipe walls. It is the special aim of this work to estimate the attenuation of pressure peaks by elastic-plastic response of the pipe walls and to calculate the remaining strains. The basic system of partial differential equations includes the conservation laws for mass and momentum and the equations of state for the fluid and the pipe wall material. Additionally, the equation of motion for the pipe wall is provided, which considers the inertia of structure response after the pressure loading. The stress is described in a one-dimensional way. Axial stresses and strains are neglected. The numerical solution of the differential equations is carried out by an explicit extended Lax-Wendroff difference-scheme with the help of an intermediate step. The accuracy of the implemented calculation models is examined using the results of experiments. First of all, pressure wave experiments are taken, where the rising pressure pulse is so chosen, that the response of the pipe wall material lies exclusively within the elastic region of the diagram. These investigations have been carried out by INTERATOM. The comparison between measured and calculated values shows an excellent agreement of experiment and calculation.

- 570 (CONF-770807--Pt.E, pp vp) LMFBR TYPICAL IMPULSIVE LOADING CHARACTERIZATION AND ITS APPLICATION TO SIMPLIFIED STRUCTURAL MODELS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The aim of this paper is a discussion of the analytical methods which have been used for the dimensioning of the support structure of a LMFBR tank to cope with the explosive accident. Particular reference is given to fast but comprehensive methods developed in order to assist the designer in the preliminary stage of the design. The three main steps necessary to define a general arrangement of the structure can be described as follows: (1) simplified equivalent characterization of the impulsive loading, (2) characterization of the elastic plastic behavior of simple structural elements, (3) parametric absorption of plastic energy with comparison with allowable strains. Each step has been extensively studied and the main results obtained in terms of the analytical methods developed and typical applications are presented and discussed. The paper is divided into two sections: analytical tools and applications.

- 571 (CONF-770807--Pt.E, pp vp) MAIN ASPECTS OF THE DESIGN OF A SUPPORT STRUCTURE OF A LMFBR WITH PARTICULAR REFERENCE TO THE EXPLOSIVE ACCIDENT CONSEQUENCES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The aim of this paper is a review of the main aspects of the design of a support structure of a LMFBR tank, with particular reference to the analysis of the non-linear dynamic behavior of the structure in the plastic range under the effect of an explosive accident within the tank. The structure is composed by a L-shaped flange, which supports the tank, connected by means of nine square beams to a rigid box-type ring, fixed to the concrete. The plug of the tank is connected to the L-shaped flange by means of a group of SS bars. The non-linear dynamic analysis of the explosive accident has been carried out on a

lumped mass model, with elastic-plastic elements which simulate main components of the support structure and tank. The impulsive load connected to the explosive accident has been modelled (on the basis of extensive comparative studies carried out) as two triangular pressure impulses has been the object of a parametric evaluation. The dynamic transient on the support structure during and after the explosive accident for each couple of pressure impulses has been analyzed by means of modified version of the NON SAP code running on a CDC 7600 computer. A large amount of results, which describe displacements, velocities and accelerations of the plug, of the tank, and of the support structure, together with the forces and stresses acting on the main structural components are presented and discussed, with particular reference to the influence of the various parameters involved in the analysis.

572 (CONF-770807--Pt.E, pp vp) CIRCUMFERENTIAL WAVE PROPAGATION EFFECTS IN THIN CYLINDRICAL VESSELS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper presents and comments results obtained from the study of an infinitely long thin cylindrical shell ($t/R=5\%$) impulsively loaded over one sixth of its perimeter. This problem is a first approach to deal with the asymmetric loading of the outer containment due to rupture of the inner vessel. A finite element approach has been adopted using the computer system EURDYN. EURDYN deals with dynamic elasto-plastic problems. Large displacements and rotations can be accounted for since a convective coordinate formulation is used. Marching in time is achieved by an explicit scheme particularly suited to the problem treated. Different material properties have been assumed: elastic-perfectly plastic and elasto-plastic with or without Bauschinger effect. In the present case simulation of the cylindrical structure is achieved by means of plane strain beam elements. In this context it is shown in details how to implement the material properties, starting from the general elasto-plastic incremental stress-strain relation for isotropic hardening. Kinematic hardening is obtained by the use of an overlay model. Although, of course, a membrane wave exists in the problem treated, a flexural wave appears; the stress amplitude of the flexural wave is about an order of magnitude above the stress amplitude of the membrane wave. This indicates that safety analysis based on axisymmetric conditions is probably not sufficient and that 3-D calculations have to be considered. Plots of the stress histories at various points are given. The analysis of the results shows that classical properties of the waves are well represented by the numerical solutions.

573 (CONF-770807--Pt.E, pp vp) RESPONSE OF SUBASSEMBLY MODEL WITH INTERNALS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

For the purpose of predicting the structural response in such accident environments, a program STRAW has been developed. This is a finite element program which can treat the structure-fluid system consisting of the coolant and the subassembly walls. Both material nonlinearities due to elastic-plastic response and geometric nonlinearities due to large displacements can be treated. The energy source can be represented either by a pressure-time history or an equation of state. Because

of the lack of any simplifying symmetry in the geometry of the subassembly the program uses a quasi-three dimensional model. The cross section of the accident hexcan and the adjacent hexcan are modelled by a two-dimensional finite element mesh which represents the hexcan walls by flexural element and the internals by two-dimensional continuum elements. This mesh is coupled to a series of one-dimensional elements which represent the axial flow of the coolant and the longitudinal stiffness of the fuel pins and hexcan. The latter is of importance in the adjacent hexcan, for its lateral displacement is resisted entirely by this flexural behavior and its inertia. The adequacy of such quasi-three dimensional models has been examined by comparing the STRAW results against almost complete three-dimensional analysis performed with the REXCAT program. In this program, the accident hexcan is represented in a true three-dimensional sense by plate-shell elements, whereas the internals are represented as axisymmetric. These comparisons indicate that the quasi-three-dimensional approach employed in STRAW is valid for a large range of pressure time histories; the fidelity of this model suffers primarily when pressure reaches a peak over a very short time, such as 5-10 microseconds.

574 (CONF-770807--Pt.E, pp vp) SUBASSEMBLY-TO-SUBASSEMBLY MELTTHROUGH ANALYTICAL STUDIES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

FUEL ASSEMBLIES;LMFBR TYPE REACTORS;MELTDOWN; S CODES

575 (CONF-770807--Pt.E, pp vp) SIMULATION OF LMFBR EXCURSION MODELS BY MEANS OF ICECO. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Some comparisons of transient fluid-structure tests with results from the ICECO containment code have already been published. The test data used before referred to simplified structural models dealing with the safety aspect of the SNR-300 fast breeder. Thus, the comparison of these more complex models with the ICECO results is the subject of this paper. The experimental data used herein pertain to test models composed of a spherical source within a cylindrical pool water. The rigid ends of the container are connected by holddown bolts and cylindrical boundaries are made of one or two concentric deformable shells. The space above the surface of the water is occupied by air which may vent during the course of the experiment. The dimensions of the model, the transient stress-strain data of the shells, which had been derived in separate experiments, and the pressure-volume relationship are known. Although the boundary conditions of the vessel are also known, they could not be simulated by the analytical models. Initially the outside shell is prestressed by the holddown bolts through the two rigid lids with rubber seals set between the ends of the shell and the lids. This rather complex boundary condition was analytically simulated in the following way: at the bottom the shells were assumed free to deform radially, but were fixed axially; at the top the shell was assumed to have no constraints at all. The analytical results show that the cylindrical vessel closest to the source begins to deform first, followed by the bottom portion of the vessel. The top vessel portion deforms only later when the fluid surface reaches the

top cover and the developed fluid pressure imparts radial vessel deformation.

576 (CONF-770807--Pt.E, pp vp) EXTENSION OF HCDA SAFETY ANALYSIS TO LARGE PCRV CONTAINMENT STRUCTURES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The prestressed concrete reactor vessel (PCRV) has thus far been applied for nuclear reactor containments involving steady-state loading. This paper, on the other hand, considers the PCRV for the LMFBR containment which must sustain dynamic effects stemming from the Hypothetical Core Disruptive Accident (HCDA). To perform a feasibility study of such a PCRV, a specialized formulation for transient analysis of such structures was developed. The analytical model of a PCRV is represented in a finite element computer code involving an explicit time-integration procedure. The formulation is particularized for the axisymmetric geometry and includes simulations of tensile cracking of concrete, reinforcement and prestressing capability. The tensile cracking of concrete and concrete reinforcement are both modeled in a single subroutine describing the element constitutive relations. The stresses in the reinforcement and concrete stresses are computed separately and combined to give an overall stress state of the composite material. The reinforcement is assumed to be elastic-plastic; the concrete is taken to be elastic in tension with a tensile stress limit. Cracking of concrete is based on the principal stress-when the tensile criterion is exceeded, a crack is assumed to form normal to the direction of the maximum principal stress. The stress component in the direction normal to the crack is then set to zero. This procedure of cracking within an element is continued until three cracks have formed, at which time the material is assumed to have lost all its tensile carrying capacity. An existing crack is also permitted to close. This option is introduced so that a compressive load can still be sustained by the material subsequent to crack formation if compressive strains develop.

577 (CONF-770807--Pt.E, pp vp) EXPERIMENTS ON THE RESPONSE OF HEXAGONAL SUBASSEMBLY DUCTS TO INTERNAL PRESSURE PULSES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The safety of Liquid Metal Fast Breeder Reactors depends in part on the ability of the hexagonal subassembly ducts (hexcans) to withstand accidental internal radial loads so that damaged hexcans can be removed from the reactor core. These internal loads could be caused by fission gas release and fuel- or steel-coolant interactions resulting from fuel-pin-cladding failure inside the hexcans. To predict deformations and strains in hexcans subjected to these internal loads and in hexcans surrounding the damaged hexcan, Argonne National Laboratory has developed the finite-element, two-dimensional structure code STRAW (Structural Transient Response of Assembly Wrappers). The experiments described here were performed to verify the modeling techniques used in STRAW and to determine the effects of material properties on hexcan response. Pressure pulses were applied in 12-inch-long hexcans of annealed, 20% CW, and 50% CW 316 stainless steel. The ends of the hexcans were fixed to ensure plane-strain response at the midsection of the hexcans. At the midsection, piezoelectric quartz pressure transducers

measured loading pressure, light-emitting-diode displacement gages measured mid-flat radial deflections, and foil strain gages measured midflat circumferential and axial strains.

578 (CONF-770807--Pt.E, pp vp) REPONSE OF WATER-FILLED PIPES TO PRESSURE PULSES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

During a hypothetical core disruptive accident (HCDA) in a fast breeder reactor, fuel meltdown in the reactor core results in high pressure molten fuel and sodium vapor expansions. These expansions accelerate the above-core sodium pool upward to impact against the vessel cover causing a large reflected pressure pulse. The pressure pulse travels downward through the sodium pool and past the outlet nozzles that lead to the primary piping loop. The pressure pulse entering the primary piping may be strong enough to plastically deform the piping and damage the intermediate heat exchanger, a part of the primary containment of the reactor. Argonne National Laboratory has developed two-dimensional, axisymmetric fluid-structure-interaction codes to predict the pressures and strains in the primary piping loop caused by HCDA-generated pressure pulses. Verification of these codes requires simple, well-defined experiments in which scale models of the primary piping are loaded with pressure pulses and the pressures, strains, and deformation in the pipe are measured. This paper presents the results of such an experimental program. Experiments were performed on 5-foot-long (3 inch OD, 0.065 inch wall) Ni 200 pipe sections filled with water. Pressure pulses from an explosively driven pulse gun had a peak pressure of about 2200 psi and a rise time of 0.125 msec and were propagated along two piping systems fixed at each pipe flange to a concrete test pad. Each piping system consisted of a pulse gun connected to a 10-foot-long, thick-walled stainless steel pipe joined rigidly to a 5-foot-long Ni 200 pipe. The Ni 200 pipe in the first piping system was terminated with a thick-walled blind flange. The Ni 200 pipe in the second piping system was joined rigidly to a second Ni 200 pipe by a thick-walled 90° elbow and was terminated by a blind flange.

579 (CONF-770807--Pt.E, pp vp) INITIAL PRESSURE SPIKE AND ITS PROPAGATION PHENOMENA IN SODIUM-WATER REACTION TESTS FOR MONJU STEAM GENERATORS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

With the objective of demonstrating the safe design of steam generators for prototype LMFBR MONJU against the postulated large-leak accident, a number of large-leak sodium-water reaction tests have been conducted using the SWAT-1 and SWAT-3 rigs. Investigation of the potential effects of pressure load on the system is one of the major concerns in these tests. This paper reports the behavior of initial pressure spike in the reaction vessel, its propagation phenomena to the simulated secondary cooling system, and the comparisons with the computer code for one-dimensional pressure wave propagation problems. Both rigs used are the scaled-down models of the helically coiled steam generators of MONJU. The SWAT-1 rig is a simplified model and consists of a reaction vessel (1/8 scale of MONJU evaporator with 0.4 m dia. and 2.5 m height) and a pressure relief system i.e., a pressure relief line and a reaction products tank. On the other hand, the SWAT-3 rig is a 1/2.5 scale

of MONJU SG system and consists of an evaporator (reaction vessel with 1.3 m dia. and 6.35 m height), a superheater, an intermediate heat exchanger (IHX), a piping system simulating the secondary cooling circuit and a pressure relief system. The both water injection systems consist of a water injection line with a rupture disk installed in front of injection hole and an electrically heated water tank. Choice of water injection rates in the scaled-down models is made based on the method of iso-velocity modeling. Test results indicated that the characteristics of the initial pressure spike are dominated by those of initial water injection which are controlled by the conditions of water heater and the size of water injection hole, etc.

- 580 (CONF-770807--Pt.E, pp vp) CRBRP STEAM GENERATOR MODELING TECHNIQUES FOR PRESSURE EXCURSIONS GENERATED BY SODIUM/WATER REACTIONS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

LEAKS; LIQUID METALS; REACTOR ACCIDENTS; SODIUM; STEAM GENERATORS; T CODES; WATER; MOLTEN METAL-WATER REACTIONS; CLINCH RIVER BREEDER REACTOR

- 581 (CONF-770807--Pt.E, pp vp) EFFECT OF LARGE SODIUM-WATER REACTIONS ON LMFBR STEAM GENERATORS: INITIAL RESULTS FROM SWAAM-1. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

EXPLOSIONS; LIQUID METALS; LMFBR TYPE REACTORS; REACTOR ACCIDENTS; S CODES; SODIUM; STEAM GENERATORS; WATER; MOLTEN METAL-WATER REACTIONS

- 582 (CONF-770807--Pt.E, pp vp) EULERIAN ALGORITHM FOR ANALYZING FLUID-STRUCTURE INTERACTION IN THE FAST REACTOR CONTAINMENT. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper presents an algorithm for analyzing the coupled fluid-structure problems in the LMFBR consisting of various internals and primary vessel. The aim is at the development of a generalized Eulerian hydrodynamic scheme, which can be used to link together the ICE technique with the finite-element structural dynamics method to provide sliding capability at interface, modeling of perforated structure, and analyses of fluid motions at the geometry discontinuity. Since the finite-element method has been already well developed, the work described herein thus will concentrate on the fluid-structure interaction. The algorithm uses the Eulerian description in the hydrodynamic formulation principally because it has the deciding advantage of treating excessive distortion. It employs the implicit temporal integration scheme to solve the nonlinear, time-dependent equations of fluid mechanics. This scheme not only ensures computational stability, but also eases the Courant condition on the time step. Furthermore, the implicit treatment of pressure and velocity fields allows the presence of all flow-speed regimes. Hence, the algorithm is applicable to any degree of excursion. Two sample problems are given, concerning wave propagation in a typical LMFBR which consists of radial shield, core barrel, core-support structure, and the primary vessel. The first problem computes the deformation histories of various structure components. The second problem examines the influence of core-support

structure openings on the pressure loading in the reactor lower plenum. Results reveal that this algorithm can offer a stable and accurate solution, and is particularly suitable for the coupled fluid-structure problems possessing strong material distortion.

- 583 (CONF-770807--Pt.E, pp vp) PARAMETRIC STUDY OF THE NONLINEAR BEHAVIOR OF COMPLETE LMFBR SUBASSEMBLIES USING THE CORTAN COMPUTER CODE. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The paper emphasizes the computational aspects of combined analytical-experimental investigations concerning dynamic elastoplastic deformation of a single fuel subassembly under transverse pressure loading. To simulate the situation within a fast reactor core during a postulated local vapor explosion the cushioning effect of the thin sodium layers between subassemblies has to be included. Therefore a new computer code CORTAN was developed which combines both a one-dimensional axial squeeze flow model as well as a variable cross-section Timoshenko beam model. The paper outlines the salient features of these computational schemes. In particular, different types of dynamic plastic material behavior formulations like the implementation of various hardening rules (e.g. isotropic, kinematic) and a newly proposed catastrophe-theoretic interpretation of strain-rate dependent dynamic unloading are discussed. As applications to fast reactor safety analysis typical results of parametric studies which show the influence of some relevant energy sources (load histories), fluid support conditions or material data on structural damage are compared with experimental findings.

- 584 (CONF-770807--Pt.F, pp vp) FROM PHENIX TO SUPER PHENIX: MECHANICAL STRUCTURES ASSURING REACTOR VESSEL TIGHTNESS AT MAIN SODIUM PUMP PENETRATIONS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Even though the components of SUPER PHENIX are larger than those of PHENIX, there is no corresponding increase in available vessel height. Furthermore the SUPER PHENIX vessel does not possess a roof. For these reasons, original and patented solutions, which, nevertheless, take advantage of the experimental results and operating experience acquired with PHENIX, were sought. The most important parameters are: weights and sizes of various components; stresses transmitted to the structures differential displacements between the top and bottom positions of components; inclinations due to deformation of the deck; preservation of containment at places where components penetrate the deck. In the solution finally adopted, an elastic torus shaped ring, which forms a ball-and-socket type joint is used. A thin shell structure connects the bottom to the reactor internals in such a way as to permit both vertical and rotational displacements. The upper joint consists of: a bolted double flange system; flanged component; a torus with a gear tooth shape profile made of elastic material; leaktight bellows attached to both flanges. Studies and calculations have been completed with the realization of a 1/5 scale model of the upper torus ring. Displacement, and angular and axial stiffness measurements have confirmed the validity of the hypotheses. Moreover, even though the lower part is an extrapolation of PHENIX, full scale experimental tests have been carried out in

sodium, reproducing the real displacements and forces that will be experienced in the reactor. Finally, the pumps will be tested with water; the suspension will be studied (investigation of relevant mechanical parameters, measurement of leaktightness with respect to the outside).

- 585 (CONF-770807--Pt.F, pp vp) EFFECTS OF SUPPORT CLAMPS ON ELEVATED TEMPERATURE PIPING. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

FASTENERS;FFTF REACTOR;FINITE ELEMENT METHOD; HIGH TEMPERATURE;PIPE FITTINGS;REACTOR COOLING SYSTEMS;STRESS ANALYSIS

- 586 (CONF-770807--Pt.L, pp vp) PROPERTY CHANGES IN FAST-REACTOR CORE-SUPPORT COMPONENTS DUE TO LOW DOSE RATE, LONG-TIME IRRADIATION. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

An analytical technique, such as the damage-function analysis, must be used to extrapolate from harder spectra data. This technique has been used to analyze the grid-support structure in the Experimental Breeder Reactor-II. The damage-function analysis has two requirements: the availability of an energy-dependent displacement cross section set for the material being investigated and the applicable neutron-energy spectra. Other investigators have developed cross sections for 18-8 stainless steels (e.g., Type 304) and the neutron-energy spectrum in EBR-II has been well characterized as a function of position during the twelve years of operation at various power levels and in different configurations. Tensile data, including yield and ultimate strengths and elongation, and swelling data are available for archive EBR-II Type 304 stainless steel irradiated for long times at moderate dose rates. Damage-function analysis was used to convert the units of neutron fluence to displacements per atom (dpa). To date, trends have been identified for residual ductility (total elongation) and swelling at the EBR-II coolant inlet temperature of 371°C. The total elongation decreases linearly with exposure from an unirradiated value of approximately 50% to approximately 28% at 2.5 dpa. Above 5 dpa, total elongation is constant at approximately 3%. Forthcoming data at 4 dpa should prove valuable in obtaining a more accurate determination of the elongation decrease with dose. Swelling of Type 304 stainless steel has been found to be almost linear with dose and independent of dose rate over the range of dose rates available. The expected doses to the EBR-II grid-support structure have been calculated as 1.5 dpa for 1976 and 3.5 dpa for 1985.

- 587 (CONF-770807--Pt.L, pp vp) EFFECTS OF STRESS CONCENTRATION AND DEFECTS ON MATERIALS CONSTITUTIVE EQUATIONS UNDER DYNAMIC LOADING. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

DEFECTS;ELASTICITY;EQUATIONS;LMFBR TYPE REACTORS;MECHANICAL TESTS;PLASTICITY;REACTOR MATERIALS;STAINLESS STEEL-304;STAINLESS STEEL-316;STAINLESS STEEL-321;STRAINS;STRESSES;STRESS ANALYSIS

- 588 (CONF-770807--Pt.L, pp vp) MECHANICAL EQUATION OF STATE ANALYSIS OF ELEVATED TEMPERATURE CYCLIC DEFORMATION OF AUSTENITIC

STAINLESS STEELS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The present work considers load relaxation data reported by Conway, Stentz and Berling, for 304 and 316 stainless steels during cyclic deformation with various hold times. The experiments were conducted at 650 and 538°C, temperatures sufficiently high so that recovery is likely to occur during the experiment. The experimental data are analyzed to determine if a power law suggested as an equation of state can represent the observed behavior. In 304 at 650°C short hold times and large cyclic strain amplitudes inhibit recovery. Conversely, long hold times and small cyclic strains promote recovery. At 538°C for 304 no evidence of recovery was found. Likewise in 316 at 650°C no recovery was found. These observations can be explained on the basis of thermal and strain induced recovery of the microstructure developed by cyclic deformation. When recovery occurs, stress relaxation no longer prescribes the loci of rate dependent behavior at a constant state; rather the material passes through a continuous succession of states. The results of this analysis indicate that the power law may provide an equation of states for the stainless steels.

- 589 (CONF-770807--Pt.L, pp vp) MICROSTRUCTURE AND LOW-CYCLE FATIGUE BEHAVIOR OF INCONEL 718. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

FATIGUE;HIGH TEMPERATURE;INCONEL 718;LMFBR TYPE REACTORS;MICROSTRUCTURE;REACTOR MATERIALS

- 590 (CONF-770817--2) OVERVIEW OF CORE DISRUPTIVE ACCIDENTS. Marchaterre, J.F. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 9p. Dep. NTIS, PC A02/MF A01.

From International seminar on containment of fast breeder reactors; San Francisco, CA, USA (22 Aug 1977).

An overview of the analysis of core-disruptive accidents is given. These analyses are for the purpose of understanding and predicting fast reactor behavior in severe low probability accident conditions, to establish the consequences of such conditions and to provide a basis for evaluating consequence limiting design features. The methods are used to analyze core-disruptive accidents from initiating event to complete core disruption, the effects of the accident on reactor structures and the resulting radiological consequences are described.

- 591 (CONF-770817--3) ASSESSMENT OF ACCIDENT ENERGETICS IN LMFBR CORE-DISRUPTIVE ACCIDENTS. Fauske, H.K. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 12p. Dep. NTIS, PC A02/MF A01.

From International seminar on containment of fast breeder reactors; San Francisco, CA, USA (22 Aug 1977).

An assessment of accident energetics in LMFBR core-disruptive accidents is given with emphasis on the generic issues of energetic recriticality and energetic fuel-coolant interaction events. Application of a few general behavior principles to the oxide-fueled system suggests that such events are highly unlikely following a postulated core meltdown event.

- 592 (CONF-770817--4) APPLICATION OF CONTAINMENT CODES TO LMFBRs IN THE UNITED STATES. Chang, Y.W. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 16p. Dep. NTIS, PC A02/MF A01.

From International seminar on containment of fast breeder reactors; San Francisco, CA, USA (22 Aug 1977).

The application of containment codes to predict the response of the fast reactor containment and the primary piping loops to HCDAs is described. Five sample problems are given to illustrate their applications. The first problem deals with the response of the primary containment to an HCDA. The second problem deals with the coolant flow in the reactor lower plenum. The third problem concerns sodium spillage and slug impact. The fourth problem deals with the response of a piping loop. The fifth problem analyzes the response of a reactor head closure. Application of codes in parametric studies and comparison of code predictions with experiments are also discussed.

- 593 (CONF-770817--5) CORE DEBRIS BEHAVIOR AND INTERACTIONS WITH CONCRETE. Baker, L. Jr. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 16p. Dep. NTIS, PC A02/MF A01.

From International seminar on containment of fast breeder reactors; San Francisco, CA, USA (22 Aug 1977).

Post-HCDA, core-debris behavior is considered in three locations: (1) within the reactor vessel, (2) within the cavity below the reactor vessel, and (3) within the concrete below the reactor cavity. In-vessel behavior is dominated by considerations of the fragmentation of molten fuel and steel on quenching in an excess of sodium to form particulate debris beds. In-vessel coolability depends primarily on the availability of sufficient horizontal surface area so that the particulate bed depth is not excessive and on the requirement that removal of small quantities of heat from the reactor vessel continues. Core debris behavior in the reactor cavity can be controlled in several different ways. An uncooled cavity liner could maintain its integrity for many days while the decay heat is absorbed by sodium boiloff from the cavity. An engineered cooling system in the cavity could result in the permanent coolability of core debris in the cavity. Such a system could be based on one of four general concepts, including (1) the cooled crucible concept, (2) the sacrificial barrier concept, (3) the stable barrier concept, and (4) the catch tray concept. Each concept has a different set of advantages and disadvantages. However, each would appear to be a viable approach to core retention.

- 594 (CONF-770822--5) OPERATING LIMITS FOR SUBASSEMBLY DEFORMATION IN EBR-II. Bottcher, J.H. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 10p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society meeting; Chattanooga, TN, USA (8 Aug 1977).

The deformation of a subassembly in response to the core environment is frequently the life limiting factor for that component in an LMFBR. Deformation can occur as diametral and axial growth or bowing of the subassembly. Such deformation has caused several handling problems in both the core and the storage basket of EBR-II and may also have contributed to reactivity anomalies during reactor operation. These problems generally affect plant availability but the reactivity anomalies

could lead to a potential safety hazard. Because of these effects the deformation mechanisms must be understood and modeled. Diametral and axial growth of subassembly ducts in EBR-II is due to swelling and creep and is a function of temperature, neutron fluence and stress. The source of stress in a duct is the hydraulic pressure difference across the wall. By coupling the calculated subassembly growth rate to the available clearance in the core or storage basket a limiting neutron fluence, or exposure, can be established.

- 595 (CONF-770822--6) EBR-II OPERATING EXPERIENCE. Smith, R.N.; Cissel, D.W.; Smith, R.R. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 52p. Dep. NTIS, PC A04/MF A01.

From American Nuclear Society meeting; Chattanooga, TN, USA (8 Aug 1977).

As originally designed and operated, EBR-II successfully demonstrated the concept of a sodium-cooled fast breeder power plant with a closed fuel reprocessing cycle (mini-nuclear park). Subsequent operation has been as an irradiation facility, a role which will continue into the foreseeable future. Since the beginning of operation in 1961, operating experience of EBR-II has been very satisfactory. Most of the components and systems have performed well. In particular, the mechanical performance of heat-removal systems has been excellent. A review of the operating experience reveals that all the original design objectives have been successfully demonstrated. To date, no failures or incidents resulting in serious in-core or out-of-core consequences have occurred. No water-to-sodium leaks have been detected over the life of the plant. At the present time, the facility is operating very well and continuously except for short shutdowns required by maintenance, refueling, modification, and minor repair. A plant factor of 76.9% was achieved for the calendar year 1976.

- 596 (CONF-770822--8) PHENIX PLANT: THREE YEARS OF OPERATION. (CEA Centre d'Etudes Nucleaires de Saclay, 91 = Gif-sur-Yvette (France); CEA Centre de Marcoule, 30 = Bagnols-sur-Geze (France); Electricite de France, 30 = Bagnols-sur-Geze. Centre de Marcoule). 1977. 14p. Dep. NTIS, MF A01.

From American Nuclear Society meeting; Chattanooga, TN, USA (8 Aug 1977).

Portions of document are illegible.

Information is presented concerning the operation and maintenance of the Phenix Reactor.

- 597 (CONF-770933--1) CRBRP FLOW INDUCED VIBRATION PROGRAM. Novendstern, E.H.; Grochowski, F.A.; Yang, T.M.; Ryan, J.A.; Mulcahy, T.M. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.; General Electric Co., Sunnyvale, CA (USA). Energy Systems and Technology Div.; Hanford Engineering Development Lab., Richland, WA (USA); Argonne National Lab., IL (USA)). 1977. Contract EY-76-C-15-2395. 57p. Dep. NTIS, PC A04/MF A01.

From International Atomic Energy Agency, international working group on fast reactors; Argonne, IL, USA (20 Sep 1977).

The program to assure the structural adequacy of CRBRP components during its planned 30 years of operation is described. The program includes (1) an assessment of reactor components relative to their susceptibility to FIV, (2) designing to minimize component excitation due to FIV, (3) scale model tests to

measure structural response during simulated operating conditions and (4) preoperational tests. An overview of the CRBRP test program is described. Additionally, details of scale model testing of reactor internals and the steam generator is described in more detail.

- 598 (CONF-770939--2) PLANT CONTROL SYSTEM DESIGN FOR THE CLINCH RIVER BREEDER REACTOR PLANT. DeMore, L.A.; Bapna, B.H.; Matta, J.V. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1977. Contract EY-76-C-15-2395. 22p. Dep. NTIS, PC A02/MF A01.

From 3. power plant dynamics, control and testing symposium; Knoxville, TN, USA (7 Sep 1977).

The design of the plant control system for the CRBR is described. The description covers both the process servos and the supervisory control system.

- 599 (CONF-770973--1) PERSPECTIVE ON LONG-RANGE NUCLEAR ENERGY OPTIONS. Harms, W.O. (Oak Ridge National Lab., TN (USA)). 1977. Contract W-7405-ENG-26. 25p. Dep. NTIS, PC A02/MF A01.

From 5. annual energy conference on energy policy options; Chicago, IL, USA (28 Sep 1977).

The study group whose effort is presented here concluded that the United States urgently needs to have a breeder option available for possible deployment before the year 2000 primarily because of uncertainties in the availability of fossil fuels and uranium supplies. It was recommended that the U/Pu LMFBR program proceed as planned, including prompt construction of the CRBRP and its associated fuel cycle facilities. Alternative cycle studies should be pursued, but without significantly delaying the current program. There are technological choices which, in suitable political contexts, may somewhat reduce proliferation risks; of these, only those that employ breeders preserve the breeder option (and the nuclear option in the long term. These alternatives must be coupled with political agreements to have any significant effect on proliferation potential internationally. These same political agreements should suffice to control the U/Pu breeder cycle; there is only a difference in degree between the U/Pu and the denatured Th/U-233 cycles.

- 600 (CONF-771008--5) RADIOLOGICAL ENVIRONMENTAL ASSESSMENT OF THE RECYCLE OF LMFBR ADVANCED FUELS. Tennery, V.J.; Bomar, E.S.; Till, J.E.; Morse, L.E.; Pobereskin, M.; Madia, W.J. (Oak Ridge National Lab., TN (USA); Battelle Columbus Labs., OH (USA)). 1977. Contract W-7405-ENG-26. 21p. Dep. NTIS, PC A02/MF A01.

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

The environmental impact resulting from the release of radioactive material during reprocessing and refabrication of spent LMFBR advanced fuels was compared with that from similar treatment of reference oxide fuel. Candidate advanced fuels include carbide [(U,Pu)C] in addition to nitride [(U,Pu)N] with selected concentrations of ^{15}N . Several techniques for preparing enriched ^{15}N were reviewed and estimates were made of the cost of preparing nitrogen enriched to greater than 99 percent by each method. Core neutronics, fuel management, and designs appropriate for each fuel were used with the DRIGEN code to calculate the compositions of spent core and

blanket fuel. The mass of fuel recycled annually was that providing 50 GW(e)-years of energy at the burnup attained by each fuel. Confinement factors for each isotope were identified for reprocessing and refabrication operations and were used to calculate source terms describing isotopic release rates. These source terms were used in the AIRDOS-II code to estimate the 50-year dose to the maximally exposed individual and to both the local and world populations. Total body dose commitments to the maximally exposed individual for oxide and carbide fuels are about 2.8 millirem, while nitride fuel would result in a range of 59 to 3.4 millirem as the ^{14}N content in fresh fuel is varied from 99.64 percent to zero.

- 601 (CONF-771023--13) ASSESSMENT OF THE PERFORMANCE OF A CONCEPTUAL ACOUSTIC SURVEILLANCE SYSTEM FOR ANOMALOUS EVENTS IN LMFBRs. Doolittle, R.D.; Carey, W.M. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 5p. Dep. NTIS, PC A02/MF A01.

From Nuclear science symposium; San Francisco, CA, USA (19 Oct 1977).

A method is developed for calculating the detectability of anomalous acoustic events. The example used is the sodium vapor bubble collapse in the subcooled regions of a Liquid Metal Fast Breeder Reactor (LMFBR). This method provides a range of estimates for detection and false alarm probabilities in acoustic surveillance systems for sodium boiling and voiding detection, as well as any other impulsive events such as loose-parts monitoring. The signal excess at the receiver array from an impulsive source is computed by an extension of methods introduced by W. Carey. Assuming an exponential pulse form for the signal (or its envelope) the equivalent source level is determined from the energy flux spectral density for inclusion in the sonar equation. The signal excess (SE) is then given by the source level (SL) minus the noise level (NL) minus the transmission loss (TL) minus the detection threshold (DT) plus the receiving array gain (AG).

- 602 (CONF-771023--22) ACOUSTIC MEASUREMENTS OF THE BOILING STABILITY TESTS ON THORS SODIUM LOOP. Sheen, S.H.; Bobis, J.P.; Carey, W.M. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 6p. Dep. NTIS, PC A02/MF A01.

From Nuclear science symposium; San Francisco, CA, USA (19 Oct 1977).

Acoustic data of boiling stability tests on the THORS (Thermal-Hydraulic Out-of-Reactor Safety) facility were obtained using three sodium-immersible high temperature microphones. The data was analyzed in both the time and frequency domains and provides the following information: (1) the acoustic signal due to sodium boiling was clearly observed; (2) the signal level and the repetition rate of boiling pulses are directly proportional to the applied heat flux; (3) a typical boiling pulse consists of a high frequency signal due mainly to the bubble collapses and a low frequency void oscillation; (4) the frequency spectra of the boiling and background pulses can be mostly assigned to various acoustic resonance frequencies of the THORS loop.

- 603 (CONF-771023--23) SIGNAL ANALYSIS METHOD USING CROSS-CORRELATION OF TURBULENCE FLOW SIGNALS TO DETERMINE CALIBRATION OF PERMANENT MAGNET SODIUM FLOWMETERS. Raptis, A.C.; Forster, G.A. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38.

6p. Dep. NTIS, PC A02/MF A01.

From Nuclear science symposium; San Francisco, CA, USA (19 Oct 1977).

Permanent magnet flowmeters are used to measure flow in sodium cooled power plants both in the core and in plant piping. In this paper the cross-correlation technique and in particular the transfer function method is used to process the flow turbulence signals from electrodes spaced along the pipe in the vicinity of the magnetic field. Results from a loop flowmeter the EBR-II secondary flowmeter and a 400 mm diameter flowmeter are presented.

604 (CONF-771036--2) DOSIMETER SAMPLE PREPARATION, QUALITY ASSURANCE, AND MATERIALS DEVELOPMENT FOR THE ILRR, FRMDC, AND FFTF DOSIMETRY PROGRAMS. Adair, H.L.; Kobisk, E.H.; Setaro, J.A.; Quinby, T.C.; Dailey, J.M. (Oak Ridge National Lab., TN (USA)). 1977. Contract W-7405-ENG-26. 19p. Dep. NTIS, PC A02/MF A01.

From 2. ASTM-EURATOM symposium on reactor dosimetry: dosimetry methods for fuels, cladding and structural materials; Palo Alto, CA, USA (2 Oct 1977).

Since the IRML dosimeter program began in 1971, approximately 1900 vanadium encapsulated dosimeters and approximately 100 aluminum encapsulated metal fission foils have been prepared to meet TLRR, FRMDC, and FFTF requirements. Development efforts have focused on cost reduction in fabrication as well as characterization of both pure and dilute materials to meet the desired accuracy goals of ± 0.5 and ± 1.0 percent (2 sigma), respectively. One important requirement of the dosimeter program is that the dosimeter materials be encapsulated in high purity vanadium. To meet this requirement over the next ten years, an adequate supply of vanadium must be maintained. Efforts were begun to locate and characterize batches of high purity vanadium to meet this requirement. In addition, efforts were initiated to obtain the necessary vanadium capsules, to be fabricated from ILRR material, on a competitive bid basis from a commercial supplier. Quality assurance procedures were established to maintain homogeneity among the various batches of dosimeters produced. Procedures developed for producing the 1.270-mm diameter capsules were adapted to the loading of the more recently required 0.889-mm diameter capsules. Vanadium capsules are to be used in neutron energy spectra, flux and fluence characterization studies during start-up and run phases of the FFTF.

605 (CONF-771052--1) MATERIAL PROPERTIES REQUIREMENTS FOR LMFBR STRUCTURAL DESIGN: GENERAL CONSIDERATIONS AND DATA NEEDS. Pugh, C.E.; Purdy, C.M. (Oak Ridge National Lab., TN (USA)); Energy Research and Development Administration, Washington, DC (USA)). 1977. Contract W-7405-ENG-26. 46p. Dep. NTIS, PC A03/MF A01.

From International working group on fast reactors (IAEA/IWGFR) specialists meeting; Bensburg, F.R. Germany (17 Oct 1977).

A statement is given of material properties information needed in connection with the structural design technology for liquid-metal fast breeder reactor (LMFBR) primary circuit components. Implementation of current analysis methods and criteria is considered with an emphasis on data and data correlations for performing elastic-plastic and creep analyses, for establishing allowable stress limits, and for computing creep-fatigue damage. Further development of the technology is discussed in relation to properties information. Emphasis is

placed on improved constitutive equations for representing inelastic material behavior, on procedures for treating time-dependent fatigue, and on criteria for creep rupture. The properties are generally discussed without regard to specific alloys, since most categories of information are needed for each major structural material. Some sample experimental results are given for type 304 stainless steel and 2 1/4 Cr-1 Mo steel.

606 (CONF-771052--2) OVERVIEW OF THE U.S. PROGRAMS ON PROPERTIES OF PRIMARY CIRCUIT MATERIALS. Brinkman, C.R.; Sikka, V.K.; Booker, M.K. (Oak Ridge National Lab., TN (USA)). 1977. Contract W-7405-ENG-26. 31p. Dep. NTIS, PC A03/MF A01.

From International working group on fast reactors (IAEA/IWGFR) specialists meeting; Bensburg, F.R. Germany (17 Oct 1977).

The objective of U.S. Breeder Reactor Programs associated with primary circuit structural materials is to develop the design data base and associated design technology on existing commercially available materials as well as new alloys. This will permit economic operation of components at acceptable levels of plant availability and at up to 40-year lifetimes for inaccessible components. Long-term component reliability, elevated-temperature service within the creep range, and resistance to sodium attack and irradiation damage, along with design in compliance with ASME Codes and RDT Specifications, have required that the U.S. programs be directed toward contributing knowledge in a number of areas. These areas, relating to material deformation, failure modes, compatibility, fabrication, long-term behavior, irradiation damage, and availability will be discussed. The U.S. Structural Material Programs concerned with primary-circuit components will be reviewed, and their current and future contributions to knowledge of these areas will be explained.

607 (CONF-771052--3) EFFECT OF SODIUM ENVIRONMENT ON THE CREEP=RUPTURE AND LOW-CYCLE FATIGUE BEHAVIOR OF AUSTENITIC STAINLESS STEELS. Natesan, K.; Chopra, J.K.; Zeman, G.J.; Smith, D.L.; Kassner, T.F. (Argonne National Lab., IL (USA)). Sep 1977. Contract W-31-109-ENG-38. 34p. Dep. NTIS, PC A03/MF A01.

From International working group on fast reactors (IAEA/IWGFR) specialists meeting; Bensburg, F.R. Germany (17 Oct 1977).

Austenitic stainless steels used for in-core structural components, piping, valves, and the intermediate heat exchanger in Liquid-Metal Fast-Breeder Reactors (LMFBRs) are subjected to sodium at elevated temperatures and to complex stress conditions. As a result, the materials can undergo compositional and microstructural changes as well as mechanical deformation by creep and cyclic fatigue processes. Information is presented on the creep-rupture and low-cycle fatigue behavior of Types 304 and 316 stainless steel in the solution-annealed condition and after long-term exposure to flowing sodium. The nonmetallic impurity-element concentrations in the sodium were controlled at levels similar to those in EBR-II primary sodium. Strain-time relationships developed from the experimental creep data were used to generate isochronous stress-creep strain curves as functions of sodium-exposure time and temperature. The low-cycle fatigue data were used to obtain relationships between plastic strain range and cycles-to-failure based on the Coffin-Manson formalism and a damage-rate approach developed at ANL. An analysis of the cyclic stress-strain

- behavior of the materials showed that the strain-hardening rates for the sodium-exposed steels were larger than those for the annealed material. However, the sodium-exposed specimens showed significant softening, as evidenced by the lower stress at half the fatigue life. Microstructural information obtained from the different specimens suggests that crack initiation is more difficult in the long-term sodium-exposed specimens when compared with the solution-annealed material. Based on the expected carbon concentrations in LMFBR primary system sodium, moderate carburization of the austenitic stainless steels will not degrade the mechanical properties to a significant extent, and therefore, will not limit the performance of out-of-core components.
- 608 (CONF-771108--3) EVALUATIONS OF CRBRP THERMAL MARGIN BEYOND THE DESIGN BASE. Schornhorst, J.R.; Strawbridge, L.E.; Bradbury, P. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1977. Contract EY-76-C-15-2395. 10p. Dep. NTIS, PC A02/MF A01.
From Post accident heat removal information exchange meeting; Chicago, IL, USA (2 Nov 1977).
The thermal loads and radiological consequences that could result from a hypothetical core disruptive accident with whole core involvement were evaluated. Since the in-vessel margins may not prevent melt-through of the reactor vessel and guard vessel, emphasis has been placed on providing margins external to the reactor vessel and guard vessel to mitigate the radiological consequences of an HCDA. To evaluate the plant thermal and radiological margins, the release of the entire core, blankets and primary sodium into the reactor cavity was considered. Design requirements have been placed on plant components and structures to assure that containment integrity can be maintained without venting for at least 24 hours. Features are also included to provide long term mitigation of HCDA consequences. Based on the thermal and radiological evaluations considering the CRBRP features it is concluded that the radiological consequences of a hypothetical core disruptive accident would be enveloped by those for light water reactors for events beyond the design base.
- 609 (CONF-771109--49) SUPPORT FOR A NUCLEAR FUTURE: STUDENT RESPONSE TO THE CRBRP. Bremseth, M.D.; Clelland, D.A. (Tennessee Univ., Knoxville (USA). Dept. of Sociology). 1977. Contract EY-76-C-15-1200. 5p. Dep. NTIS, PC A02/MF A01.
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
Results are presented of a multiple regression analysis of questionnaire data from two random samples of University of Tennessee seniors. Data were collected from 94 students with science/engineering majors (the 'Tech' sample), and 91 students with non-science/engineering majors (the 'Mass' sample--which represents the majority of the students). The purpose of the analysis was to isolate factors which independently explain student response to the CRBRP and to breeder reactors in general.
- 610 (CONF-771109--52) CONCEPTS OF REMOTE SIGNAL TRANSMISSION FOR LIQUID-METAL FAST BREEDER REACTORS. De Lorenzo, J.T.; Burns, R.S.; Valentine, K.H. (Oak Ridge National Lab., TN (USA)). 1977. Contract W-7405-ENG-26. 4p. Dep. NTIS, PC A02/MF A01.
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
Sensors and remote signal transmission techniques suitable for LMFBR conditions are discussed. Some of the major problems which must be addressed during further development of sensors and techniques are listed.
- 611 (CONF-771109--77) DYNAMIC BOILING TESTS IN A 19-PIN SIMULATED LMFBR FUEL ASSEMBLY. Wantland, J.L.; Clapp, N.E.; Fontana, M.H.; Gnadt, P.A.; Hanus, N. (Oak Ridge National Lab., TN (USA)). 1977. Contract W-7405-ENG-26. 22p. Dep. NTIS, PC A02/MF A01.
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
A series of quasi-steady state tests were made to study dynamic boiling and to determine a boiling region which would not lead to immediate dryout and pin failure in the Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility (previously the Fuel Failure Mockup), an engineering-scale sodium loop for testing simulated LMFBR fuel assemblies at normal and off-normal operating conditions. These tests were made on an assembly consisting of 19 electrically heated pins of the same configuration as FFTF fuel. Approximately 50 flow transients were so conducted to boiling or near boiling over a range of flows and powers. Dryout occurred at three test conditions.
- 612 (CONF-771109--78) MAXIMUM WAKE TEMPERATURE AND NUSSELT NUMBER BEHIND BLOCKAGES IN SODIUM-COOLED BUNDLES. Han, J.T.; Fontana, M.H.; Wantland, J.L. (Oak Ridge National Lab., TN (USA)). 1977. Contract W-7405-ENG-26. 5p. Dep. NTIS, PC A02/MF A01.
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
Two important correlations have been obtained to calculate the maximum coolant temperature and the Nusselt number in the wake (the flow recirculating zone) downstream of blockages in 19-pin sodium-cooled bundles. These correlations can be applied to predict the maximum temperature rise and the average heat transfer coefficient behind small non-heat-generating blockages in the fuel assemblies of the FFTF and the CRBR for the wide range of flow and power conditions. Experiments with partial blockages in simulated LMFBR fuel assemblies have been performed at the THORS facility (formerly called FFM) of the Oak Ridge National Laboratory. Nineteen-pin sodium-cooled bundles were used, which had the same pin diameter and pitch as the CRBR and the FFTF fuel assemblies.
- 613 (CONF-771109--86) DESIGN AND DEVELOPMENT OF THE CRBRP EX-VESSEL TRANSFER MACHINE. Jones, C.E. Jr. (Atomic International Div., Canoga Park, CA (USA)). 1977. 24p. Dep. NTIS, PC A03/MF A01.
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
The Reactor Refueling System (RRS) for the Clinch River Breeder Reactor Project (CRBRP) uses the Ex-Vessel Transfer Machine (EVTM) for transferring core assemblies outside the reactor vessel. The design of the Ex-Vessel Transfer Machine (EVTM) and its gantry-trolley for the CRBRP is discussed. The development tests required for the design are presented, in conjunction with the impact of the test results on the design. The impact of the increased seismic requirements on the design are also presented.
- 614 (CONF-771120--8) EDGE BLOCKAGE IN A 19-PIN SODIUM-COOLED BUNDLE. Han, J.T.; Gnadt,

P.A.; Fontana, M.H.; Wantland, J.L. (Oak Ridge National Lab., TN (USA)). 1977. Contract W-7405-ENG-26. 9p. Dep. NTIS, PC A02/MF A01.

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

To assess the effect of partial flow blockages on the local temperature distributions in LMFBR fuel assemblies, several experiments have been carried out at the Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility at the Oak Ridge National Laboratory. Two kinds of flow blockages have been investigated: (1) six-channel internal blockage for which the blocked channels are not adjacent to the duct wall, and (2) 14-channel edge blockage for which some of the blocked channels are adjacent to the wall. The larger edge blockage is generally more severe than the internal blockage due to higher temperature rises in the wake behind the blockage. The experimental results on the edge blockage are analyzed and presented.

- 615 (CONF-771120--14) BLOCKAGES IN LMFBR FUEL ASSEMBLIES: A REVIEW. Han, J.T.; Fontana, M.H. (Oak Ridge National Lab., TN (USA)). 1977. Contract W-7405-ENG-26. 73p. Dep. NTIS, PC A03/MF A01.

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

Experimental and analytical investigations performed in the United States, Germany, Great Britain, and Japan on the effects of partial flow blockages in liquid-metal fast breeder reactor fuel assemblies are reviewed and the results presented. Generalized models are developed from experimental data obtained for blockages of various sizes, shapes, and porosity, with and without pins, utilizing water and sodium as the coolant. Generally, the recirculating flow in the wake behind a blockage is a relatively effective heat transfer mechanism. Experiments where sodium boiling was made to occur behind the blockages indicate that boiling is stable for the configurations tested; these results are predicted by analytical models. Blockages at the inlet of fuel assemblies tend to have insignificant effects in the fuel assembly unless flow is reduced grossly and therefore would be detectable. Blockages in the heat generating zone have to be quite large to cause sodium boiling under normal reactor operating conditions.

- 616 (CONF-771120--20) COMPUTING THE EFFECT OF PLASTIC DEFORMATION OF PIPING ON PRESSURE TRANSIENT PROPAGATION. Youngdahl, C.K.; Kot, C.A. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 15p. Dep. NTIS, PC A02/MF A01.

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

The computer program PTA-1 performs pressure-transient analysis of large piping networks using the one-dimensional method of characteristics applied to a fluid-hammer formulation. The effect of elastic-plastic deformation of piping on pulse propagation is included in the computation. Each pipe is modeled as a series of rings, neglecting axial effects, bending moments, and inertia. The fluid wave speed is a function of pipe deformation and, consequently, of position and time. Comparison with existing experimental data indicate that this simple fluid-structure interaction model gives surprisingly accurate results for both pressure histories in the

fluid and strain histories in the piping.

- 617 (CONF-771120--22) LIQUID METAL FAST BREEDER REACTOR SAFETY: AN OVERVIEW INCLUDING IMPLICATIONS OF ALTERNATE FUEL CYCLES. Fauske, H.K. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 33p. Dep. NTIS, PC A03/MF A01.

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

Key issues concerning LMFBR safety are reviewed. A brief discussion of alternate fuel cycles for preventing diversion of fissionable materials is included.

- 618 (CONF-771120--23) EULERIAN METHOD FOR LARGE-DISPLACEMENT FLUID-STRUCTURE INTERACTION IN REACTOR CONTAINMENTS. Chang, Y.W.; Wang, C.Y. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 17p. Dep. NTIS, PC A02/MF A01.

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

An Eulerian method for analyzing large-displacement fluid-structure interaction in reactor containments is presented. The emphasis is on the development of a generalized hydrodynamic scheme to treat the irregular cells created by the movement of the structure with respect to the fixed Eulerian coordinates. A relaxation equation is derived from the boundary condition at the fluid-structure interface for the solution of the pressure at the interface. By combining this with the Poisson equation a sufficient set of equations is obtained for the determination of the advanced-time pressures in the fluid region and at the fluid-structure interfaces. Sample problems are given to illustrate the analysis.

- 619 (CONF-771123--1) ENGINEERING ASPECTS OF HETEROGENEOUS AND HOMOGENEOUS REACTORS. Dickson, P.W. Jr. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1977. Contract EY-76-C-15-2395. 27p. Dep. NTIS, PC A03/MF A01.

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The core arrangement in an LMFBR can potentially affect the lower internals, upper internals, radial shielding, vessel, hot leg transients, head access area and control systems as well as breeding ratio, doubling time and core inventory. This paper describes the results of a study of the impact on these components and parameters that would result if the Clinch River Breeder Reactor were to incorporate a heterogeneous core.

- 620 (CONF-771123--2) POTENTIAL OF LARGE HETEROGENEOUS REACTORS. Barthold, W.P.; Tzanos, C.P.; Beitel, J.C. (Argonne National Lab., IL (USA)). 1977. Contract W-31-109-ENG-38. 24p. Dep. NTIS, PC A02/MF A01.

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

Different 1200 MWe heterogeneous core configurations have been evaluated and compared with a homogeneous 1200 MWe LMFBR design. This study discusses the neutronic coupling concept which was employed to discriminate between different heterogeneous core designs. Reactivity control, breeding performance, safety parameters, fuel pin optimization, transient behavior and thermal performance were investigated in detail and results of these

analyses are presented.

LMFBR TYPE REACTORS; DENATURED FUEL; FUEL CYCLE; COMPARATIVE EVALUATIONS

- 621 (CONF-771155--P2, pp 327-348) ECONOMIC REGIMES. Deonigi, D.E. (Battelle Pacific Northwest Labs., Richland, WA). Jul 1978.
From 2. DMFE hybrid reactor meeting; Washington, DC, USA (2 Nov 1977).
Cost-benefit studies were made for LWR's, LMFBR, CTR, hybrid reactors, and coal. Fuel cost figures are given. Forecasts were made for the year 2000. (MOW)
- 622 (CONF-771203--, pp 523-525) MATERIALS FOR ADVANCED NUCLEAR ENERGY SYSTEMS. Steele, L.E. (Naval Research Lab., Washington, DC). 1977.
From Alternative energy sources symposium; Miami Beach, FL, USA (5 Dec 1977).
PWR TYPE REACTORS; BWR TYPE REACTORS; LMFBR TYPE REACTORS; THERMONUCLEAR REACTORS; REACTOR MATERIALS; PHYSICAL RADIATION EFFECTS
- 623 (CONF-771203--, pp 113-115) LMFBR: LOGIC AND TECHNOLOGY. Little, W.W. (Hanford Engineering Development Lab., Richland, WA). 1977.
From Alternative energy sources symposium; Miami Beach, FL, USA (5 Dec 1977).
LMFBR TYPE REACTORS; FUEL CYCLE; ECONOMICS; USA; URANIUM RESERVES
- 624 (CONF-771203--, pp 117-119) MEETING U.S. ELECTRICAL NEEDS: IS THE BREEDER URGENT. Osborne, T.C. (General Electric Co., Sunnyvale, CA); Palmer, R.S.; Wyckoff, H.L. 1977.
From Alternative energy sources symposium; Miami Beach, FL, USA (5 Dec 1977).
USA; NUCLEAR POWER; LMFBR TYPE REACTORS; ENERGY SUPPLIES; ELECTRIC POWER; PLANNING; ENERGY POLICY
- 625 (CONF-771203--, pp 121-122) ROLE OF THE BREEDER REACTOR IN OUR ENERGY FUTURE. Rop, D.J. (Westinghouse Electric Corp., Madison, PA). 1977.
From Alternative energy sources symposium; Miami Beach, FL, USA (5 Dec 1977).
USA; LMFBR TYPE REACTORS; ENERGY POLICY; PLANNING
- 626 (CONF-771203--, pp 127-129) THERMAL ANALYSIS OF A VARIABLE GEOMETRY MULTI-REGION NUCLEAR FUEL ROD. Almond, M.D.; Wolf, H. (Univ. of Arkansas, Fayetteville). 1977.
From Alternative energy sources symposium; Miami Beach, FL, USA (5 Dec 1977).
FUEL RODS; FUEL PELLETS; HEAT TRANSFER; URANIUM DIOXIDE; PLUTONIUM DIOXIDE; LMFBR TYPE REACTORS
- 627 (CONF-771203--, pp 131-133) EBR-II EXPERIENCE ON FISSION-PRODUCT RELEASE FROM SODIUM-BONDED FUEL ELEMENTS. So, B.Y.C.; Lambert, J.D.B. (Argonne National Lab., IL). 1977.
From Alternative energy sources symposium; Miami Beach, FL, USA (5 Dec 1977).
EBR-2 REACTOR; FUEL ELEMENTS; FISSION PRODUCT RELEASE; FUEL CANS
- 628 (CONF-771203--, pp 491-495) ALTERNATIVE FUEL CYCLES FOR BREEDER REACTORS. Burns, T.J. (Oak Ridge National Lab., TN); Bartine, D.E.; White, J.R.; Kallfelz, J.M.; Rowland, D.M. 1977.
From Alternative energy sources symposium; Miami Beach, FL, USA (5 Dec 1977).
- 629 (CONF-771203--, pp 497-499) USE OF ^{232}Th - ^{233}U FUEL CYCLE IN FAST BREEDER REACTORS. Horak, J.A.; Lotts, A.L. (Oak Ridge National Lab., TN). 1977.
From Alternative energy sources symposium; Miami Beach, FL, USA (5 Dec 1977).
LMFBR TYPE REACTORS; THORIUM CYCLE; FUEL CYCLE
- 630 (CONF-771203--, pp 601) ROAD TO ENERGY INDEPENDENCE: THE BREEDER REACTOR. Anderson, J.C. (Oak Ridge Associated Universities, TN). 1977.
From Alternative energy sources symposium; Miami Beach, FL, USA (5 Dec 1977).
LMFBR TYPE REACTORS; CLINCH RIVER BREEDER REACTOR; PLANNING; ENERGY POLICY
- 631 (CONF-771203--, pp 123-125) INHERENT EMERGENCY COOLING OF LIQUID METAL FAST BREEDER REACTORS BY NATURAL CIRCULATION. Singer, R.M.; Mohr, D. (Argonne National Lab., IL). 1977.
From Alternative energy sources symposium; Miami Beach, FL, USA (5 Dec 1977).
LMFBR TYPE REACTORS; REACTOR ACCIDENTS; ECCS
- 632 (CONF-771203--, pp 135-137) REACTOR SCRAM EXPERIENCE FOR RELIABILITY ANALYSIS OF LMFBR SHUTDOWN SYSTEMS. Gerstner, M.T.; Edison, G.E. (Westinghouse Electric Corp., Madison, PA). 1977.
From Alternative energy sources symposium; Miami Beach, FL, USA (5 Dec 1977).
LMFBR TYPE REACTORS; SCRAM; REACTOR PROTECTION SYSTEMS; RELIABILITY
- 633 (CONF-771203--, pp 773-774) CRBRP SAFETY STUDY: AN ASSESSMENT OF RISK. Piper, H.B.; Conradi, L.L.; Buhl, A.R.; Wood, P.J.; Leaver, D.E.W. 1977.
From Alternative energy sources symposium; Miami Beach, FL, USA (5 Dec 1977).
CLINCH RIVER BREEDER REACTOR; REACTOR SAFETY; RISK ASSESSMENT
- 634 (CONF-771217--1) CRBRP DESIGN AND TEST RESULTS FOR FUEL HANDLING SYSTEMS, PLUGS, AND SEALS. Berg, G.E. (Atomics International Div., Canoga Park, CA (USA)). 1977.
Contract EY-76-C-15-1200. 20p. Dep. NTIS, PC A02/MF A01.
From Seminar on LMFBR components; Canoga Park, CA, USA (5 Dec 1977).
The fuel handling system and reactor rotating plugs for the Clinch River Breeder Reactor Plant (CRBRP) are based primarily on existing technology and, in many respects, follow the concept developed for the Fast Flux Test Facility (FFTF). The equipment and the development programs initiated to verify its performance are described. Test results obtained from the development program, and the extent to which these results verified original design selections, or suggested potential improvements, are discussed.
- 635 (CONF-771217--2) FFTF AND CRBRP INTERMEDIATE HEAT EXCHANGER DESIGN, TESTING, AND FABRICATION. Devlin, R.W.; Bresnahan, J.D. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1977.
Contract EY-76-C-02-3045-022. 34p. Dep. NTIS, PC A03/MF A01.
From Seminar on LMFBR components; Canoga Park, CA, USA (5 Dec 1977).

The Heat Transport Systems of the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor Plant (CRBRP) include Intermediate Heat Exchangers (IHxs) to form barriers between the primary radioactive and secondary non-radioactive sodium coolant loops. The design, development and fabrication of the IHX for the FFTF has been completed and the units installed in the facility. The design and development of the IHX for the CRBRP has progressed through the performance sizing stage and flow testing of critical areas into early fabrication. The design evolution of these two heat exchangers may be considered a continuous development because the design and fabrication organizations have been the same for both units. Structural considerations to meet ASME Code requirements, support testing experience and manufacturing experience from the FFTF program have been factored into the CRBRP program. This paper outlines the similarities and differences between the FFTF and CRBRP designs. Special attention has been given to utilization of flow distribution devices and associated hydraulic testing, tube bundle and tube-to-tubesheet joint configurations, and fabrication differences.

- 636 (CONF-771217--3) CLINCH RIVER BREEDER REACTOR SECONDARY CONTROL ROD SYSTEM. McKeehan, E.R.; Sim, R.G. (Energy, Inc., Idaho Falls, ID (USA)). 14 Sep 1977. Contract EY-76-C-02-3045-002. 19p. Dep. NTIS, MF A01.

From Seminar on LMFBR components; Canoga Park, CA, USA (5 Dec 1977).

Portions of document are illegible.

The shutdown system for the Clinch River Breeder Reactor (CRBR) includes two independent systems--a primary and a secondary system. The Secondary Control Rod System (SCRS) is a new design which is being developed by General Electric to be independent from the primary system in order to improve overall shutdown reliability by eliminating potential common-mode failures. The paper describes the status of the SCRS design and fabrication and testing activities. Design verification testing on the component level is largely complete. These component tests are covered with emphasis on design impact results. A prototype unit has been manufactured and system level tests in sodium have been initiated.

- 637 (CONF-771217--4) CRBRP SODIUM CIRCULATING PUMP DESIGN EVALUATION. Marrujo, F.; Cook, M.; Manners, L.; Cothran, H. (Clinch River Breeder Reactor Plant Project Office, Oak Ridge, TN (USA)). Dec 1977. Contract EY-76-C-02-3045-014. 26p. Dep. NTIS, PC A03/MF A01.

From Seminar on LMFBR components; Canoga Park, CA, USA (5 Dec 1977).

The following topics are discussed: (1) primary sodium pump design concept; (2) pump level control system; (3) resolution of design problems in stress analysis, dynamics analysis, and mechanical design; (4) model testing; (5) planned performance tests; and (6) fabrication status. (DG)

- 638 (CONF-771217--5) REVIEW OF FFTF AND CRBRP CONTROL ROD SYSTEMS DESIGNS. Pitterle, T.A.; Legally, H.O. (Clinch River Breeder Reactor Plant Project Office, Oak Ridge, TN (USA)). 4 Oct 1977. Contract EY-76-C-02-3045-002. 30p. Dep. NTIS, PC A02/MF A01.

From Seminar on LMFBR components; Canoga Park, CA, USA (5 Dec 1977).

The evolution of the primary control rod system design for FFTF and CRBR, beginning with

the initial choice of the basic concepts, is described. The significant component and systems tests are reviewed together with the test results which referenced the development of the CRBR primary control rod system design. Modifications to the concepts and detail designs of the FFTF control rod system were required principally to satisfy the requirements of CRBR, and at the same time incorporating design refinements shown desirable by the tests.

- 639 (CONF-771217--6) CRBRP DECAY HEAT REMOVAL SYSTEMS. Hottel, R.Z.; Louison, R.; Boardman, C.E.; Kiley, M.J. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.; General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.; Atomics International Div., Canoga Park, CA (USA)). 1977. Contract EY-76-C-02-3045. 31p. Dep. NTIS, PC A03/MF A01.

From Seminar on LMFBR components; Canoga Park, CA, USA (5 Dec 1977).

The Decay Heat Removal Systems for the Clinch River Breeder Reactor Plant (CRBRP) are designed to adequately remove sensible and decay heat from the reactor following normal shutdown, operational occurrences, and postulated accidents on both a short term and a long term basis. The Decay Heat Removal Systems are composed of the Main Heat Transport System, the Main Condenser and Feedwater System, the Steam Generator Auxiliary Heat Removal System (SGAHS), and the Direct Heat Removal Service (DHRS). The overall design of the CRBRP Decay Heat Removal Systems and the operation under normal and off-normal conditions is examined. The redundancies of the system design, such as the four decay heat removal paths, the emergency diesel power supplies, and the auxiliary feedwater pumps, and the diversities of the design such as forced circulation/natural circulation and AC Power/DC Power are presented. In addition to overall design and system capabilities, the detailed designs for the Protected Air Cooled Condensers (PACC) and the Air Blast Heat Exchangers (ABHX) are presented.

- 640 (CONF-771217--7) DESIGN OF THE CORE SUPPORT AND RESTRAINT STRUCTURES FOR FFTF AND CRBRP. Sutton, H.G.; Rylatt, J.A. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Dec 1977. Contract EY-76-C-02-3045-005. 25p. Dep. NTIS, PC A02/MF A01.

From Seminar on LMFBR components; Canoga Park, CA, USA (5 Dec 1977).

This paper presents and compares the design and fabrication of the FFTF and CRBRP reactor structures which support and restrain the reactor core assemblies. The fabrication of the core support structure (CSS) for the FFTF reactor was completed October 1972 and this paper discusses how the fabrication problems encountered with the FFTF were avoided in the subsequent design of the CRBR CSS. The radial core restraint structure of the FFTF was designed and fabricated such that an active system could replace the present passive system which is segmented and relies on the CSS core barrel for total structure integrity to maintain core geometry. The CRBR core restraint structure is designed for passive restraint only, and this paper discusses how the combined strengths of the restraint structure former rings and the CSS core barrel are utilized to maintain core geometry. Whereas the CSS for the FFTF interfaces directly with the reactor core assemblies, the CRBR CSS does not. A comparison is made on how intermediate structures in CRBR (inlet modules) provide the necessary design

interfaces for supporting and providing flow distribution to the reactor core assemblies. A discussion is given on how the CRBR CSS satisfied the design requirements of the Equipment Specification, including thermal transient, dynamic and seismic loadings, and results of flow distribution testing that supported the CRBR design effort. The approach taken to simplify fabrication of the CRBR components, and a novel 20 inch deep narrow gap weld joint in the CSS are described.

- 641 (CONF-771217--8) DESIGN OF THE UPPER INTERNALS STRUCTURE FOR THE CLINCH RIVER BREEDER REACTOR PLANT. Thompson, D.C.; Novendstern, E.H. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1977. Contract EY-76-C-02-3045. 30p. Dep. NTIS, PC A03/MF A01.

From Seminar on LMFBR components; Canoga Park, CA, USA (5 Dec 1977).

The Upper Internals Structure (UIS) is located above the core and is supported from the head at four locations. It is designed to perform the following primary functions: provide secondary core holddown in the event of a malfunction of the core hydraulic holddown system; provide support for routing all in-vessel instrumentation to core assemblies; maintain alignment between the core assemblies, the UIS and the closure head; provide guidance and crossflow protection for the control rod drivelines; and mix/duct flow to the upper region of the vessel outlet plenum to minimize rapid temperature changes to components during a reactor trip transient. In accomplishing these functions, the UIS will experience a sodium environment with temperatures up to 1200°F (649°C), and as many as 7×10^6 cycles of fluid temperature fluctuations up to 250°F (121°C) at full power operation. It must be designed to survive these conditions in combination with seismic and flow-induced vibration loadings for its 30 year design life. The design program of designing to controlled functional requirements and design conditions is discussed. Included is a description of the significant parts of the design and the approach used to balance the requirement of tight joints. The thermal and hydraulic environment including the results of a comprehensive test program are discussed. The test program results establish the basis of the thermal boundary used in the structural evaluation, and the UIS vibration characteristics. A summary of the areas which have required design changes is included with a summary of the structural evaluation of these changes.

- 642 (CONF-771217--9) FFTF AND CRBRP REACTOR VESSELS. Morgan, R.E. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1977. Contract EY-76-C-02-3045-002. 12p. Dep. NTIS, PC A02/MF A01.

From Seminar on LMFBR components; Canoga Park, CA, USA (5 Dec 1977).

The Fast Flux Test Facility (FFTF) reactor vessel and the Clinch River Breeder Reactor Plant (CRBRP) reactor vessel each serve to enclose a fast spectrum reactor core, contain the sodium coolant, and provide support and positioning for the closure head and internal structure. Each vessel is located in its reactor cavity and is protected by a guard vessel which would ensure continued decay heat removal capability should a major system leak develop. Although the two plants have significantly different thermal power ratings, 400 megawatts for FFTF and 975 megawatts for CRBRP, the two reactor vessels are comparable in size, the CRBRP vessel being approximately

28% longer than the FFTF vessel. The FFTF vessel diameter was controlled by the space required for the three individual In-Vessel Handling Machines and Instrument Trees. Utilization of the triple rotating plug scheme for CRBRP refueling enables packaging of the larger CRBRP core in a vessel the same diameter as the FFTF vessel.

- 643 (CONF-780203--9) U.S. PROGRAM FOR LARGE SODIUM/WATER REACTION TESTS. Whipple, J.C.; Amos, J.C.; DeBear, W.S.; Neely, H.H.; Hopenfeld, J. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). [nd]. Contract EY-76-C-03-0189; EY-76-C-03-0701. 26p. Dep. NTIS, PC A03/MF A01.

From 1. joint U.S./Japan LMFBR steam generator seminar; Tokyo, Japan (13 Feb 1978).

An extensive large leak test program is underway in the United States to investigate the effects of large sodium-water reaction events in LMFBR steam generators. The general objectives of the program are to provide data for validating analytical codes, provide information on the extent of secondary tube failures, provide data on relief system performance and provide data on cleanup and recovery needs. Tests are conducted in the Large Leak Test Rig located at the Liquid Metal Engineering Center. The program is divided into two phases, Series I and Series II, for the purpose of satisfying near-term and long-term needs. The Series I testing, utilizing the Atomics International modular steam generator as the test article, commenced in July 1976 with six tests completed by November 1977. The test article for the Series II tests (the Large Leak Test Vessel/Large Leak Test Internals) has been fabricated. Upon completion of Large Leak Test Rig modification, test article installation, and system checkout Series II testing will begin in the Spring of 1979. The overall program plan is defined, the test articles and test facility are described, and the experimental approach is discussed for both Series I and II programs. The Series I test experience and results of intertest examinations are also presented.

- 644 (CONF-780215--2) NEEDS FOR DEVELOPMENT IN NONDESTRUCTIVE TESTING FOR ADVANCED REACTOR SYSTEMS. McClung, R.W. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 23p. Dep. NTIS, PC A02/MF A01.

From Conference on NDE in the nuclear industry; Salt Lake City, UT, USA (13 Feb 1978).

The needs for development of nondestructive testing (NDT) techniques and equipment were surveyed and analyzed relative to problem areas for the Liquid-Metal Fast Breeder Reactor, the Molten-Salt Breeder Reactor, and the Advanced Gas-Cooled Reactor. The paper first discusses the developmental needs that are broad-based requirements in nondestructive testing, and the respective methods applicable, in general, to all components and reactor systems. Next, the requirements of generic materials and components that are common to all advanced reactor systems are examined. Generally, nondestructive techniques should be improved to provide better reliability and quantitativeness, improved flaw characterization, and more efficient data processing. Specific recommendations relative to such methods as ultrasonics, eddy currents, acoustic emission, radiography, etc., are made. NDT needs common to all reactors include those related to materials properties and degradation, welds, fuels, piping, steam

generators, etc. The scope of applicability ranges from initial design and material development stages through process control and manufacturing inspection to in-service examination.

- 645 (CONF-780232--1) EBR-II EXPERIENCE WITH SODIUM CLEANING AND RADIOACTIVITY DECONTAMINATION. Ruther, W.E.; Smith, C.R.F. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 27p. Dep. NTIS, PC A03/MF A01.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

The EBR-2 is now in its 13th year of operation. During that period more than 2400 subassemblies have been cleaned of sodium without a serious incident of any kind by a two-step process developed at Argonne. Sodium cleaning and decontamination of other reactor components has been performed only on the relatively few occasions in which a repair or replacement has been required. A summary of the EBR-2 experience will be presented. A new facility will be described for the improved cleaning and maintenance of sodium-wetted primary components.

- 646 (CONF-780251--P2) PROCEEDINGS OF THE U.S./U.S.S.R. SEMINAR ON PROBLEMS OF DESIGN, DEVELOPMENT, FABRICATION, AND TEST OF BREEDER REACTOR COMPONENTS. (Department of Energy, Washington, DC (USA). Div. of Reactor Research and Technology). Aug 1978. 387p. AT.

From US/USSR seminar on problems of design, development, fabrication, and test of breeder reactor components; Los Angeles, CA, USA (6 Feb 1978).

Each paper has been separately processed for the Energy Information Data Base.

- 647 (CONF-780251--P2, pp 1.1-1.12) FFTF AND CRBRP REACTOR VESSELS. Morgan, R.E. Aug 1978.

From US/USSR seminar on problems of design, development, fabrication, and test of breeder reactor components; Los Angeles, CA, USA (6 Feb 1978).

Included are design specifications, descriptions, and detailed illustrations of FFTF and CRBRP reactor vessels and associated structures.

- 648 (CONF-780251--P2, pp 2.13-2.36) DESIGN OF THE CORE SUPPORT AND RESTRAINT STRUCTURES FOR FFTF AND CRBRP. Sutton, H.G.; Rylatt, J.A. Aug 1978.

From US/USSR seminar on problems of design, development, fabrication, and test of breeder reactor components; Los Angeles, CA, USA (6 Feb 1978).

The paper presents and compares the design and fabrication of the FFTF and CRBRP reactor structures which support and restrain the reactor core assemblies. The fabrication of the core support structure (CSS) for the FFTF reactor was completed October 1972 and the paper discusses how the fabrication problems encountered with the FFTF were avoided in the subsequent design of the CRBR CSS. The radial core restraint structure of the FFTF was designed and fabricated such that an active system could replace the present passive system which is segmented and relies on the CSS core barrel for total structure integrity to maintain core geometry. The CRBR core restraint structure is designed for passive restraint only, and the paper discusses how the combined strengths of the restraint structure former

rings and the CSS core barrel are utilized to maintain core geometry. Whereas the CSS for the FFTF interfaces directly with the reactor core assemblies, the CRBR CSS does not. A comparison is made on how intermediate structures in CRBR (inlet modules) provide the necessary design interfaces for supporting and providing flow distribution to the reactor core assemblies. A discussion is given on how the CRBR CSS satisfied the design requirements of the Equipment Specification, including thermal transient, dynamic and seismic loadings, and results of flow distribution testing that supported the CRBR design effort. The approach taken to simplify fabrication of the CRBR components, and a novel 20 inch deep narrow gap weld joint in the CSS are described.

- 649 (CONF-780251--P2, pp 3.37-3.56) CRBRP DESIGN AND TEST RESULTS FOR FUEL HANDLING SYSTEMS, PLUGS, AND SEALS. Berg, G.E. Aug 1978.

From US/USSR seminar on problems of design, development, fabrication, and test of breeder reactor components; Los Angeles, CA, USA (6 Feb 1978).

The fuel handling system and reactor rotating plugs for the Clinch River Breeder Reactor Plant (CRBRP) are based primarily on existing technology and, in many respects, follow the concept developed for the Fast Flux Test Facility (FFTF). However, several requirements are considerably more stringent and demand more advanced or larger equipment. The paper first briefly describes the equipment, and then concentrates on the development programs initiated to verify its performance. Tests results obtained from the development program, and the extent to which these results verified original design selections, or suggested potential improvements, are discussed. Among the designs (and tests) included are the In-Vessel Transfer Machine (component and full-size prototype tests), Reactor Rotating Plugs (dynamic inflatable seal, sodium dip seal, and functional test), Ex-Vessel Transfer Machine (heat transfer and component tests), Ancillary Components (core component pot siphon and rotating guide tube tests), and Fuel Handling Cell (component operability and operator training).

- 650 (CONF-780251--P2, pp 4.57-4.85) DESIGN OF THE UPPER INTERNALS STRUCTURE FOR THE CLINCH RIVER BREEDER REACTOR PLANT. Thompson, D.C.; Novendstern, E.H. Aug 1978.

From US/USSR seminar on problems of design, development, fabrication, and test of breeder reactor components; Los Angeles, CA, USA (6 Feb 1978).

The Upper Internals Structure (UIS) is located above the core and is supported from the head at four locations. It is designed to perform the following primary functions: provide secondary core holddown in the event of a malfunction of the core hydraulic holddown system; provide support for routing all in-vessel instrumentation to core assemblies; maintain alignment between the core assemblies, the UIS and the closure head; provide guidance and crossflow protection for the control rod drive lines; and mix/duct flow to the upper region of the vessel outlet plenum to minimize rapid temperature changes to components during a reactor trip transient. In accomplishing these functions, the UIS will experience a sodium environment with temperatures up to 1200°F (649°C), and as many as 7×10^6 cycles of fluid temperature fluctuations up to 250°F (121°C) at full power operation. It must be designed to survive these conditions in

combination with seismic and flow-induced vibration loadings for its 30 year design life. The paper reflects the design progress of designing to controlled functional requirements and design conditions. Included is a description of the significant parts of the design and the approach used to balance the requirement of tight joints. The paper includes a discussion of the thermal and hydraulic environment including the results of a comprehensive test program. The test program results establish the basis of the thermal boundary used in the structural evaluation, and the UIS vibration characteristics. A summary of the areas which have required design changes is included with a summary of the structural evaluation of these changes.

- 651 (CONF-780251--P2, pp 5.87-5.106) DESIGN AND OPERATING EXPERIENCE OF EBR-II INTERMEDIATE HEAT EXCHANGER (IHX). Buschman, H.W.; Koenig, J.F.; Stone, C.C. Aug 1978.

From US/USSR seminar on problems of design, development, fabrication, and test of breeder reactor components; Los Angeles, CA, USA (6 Feb 1978).

The IHX assembly consists of three basic sections: (a) the well casing, (b) the tube bundle, and (c) the shield plug. The well casing is permanently installed within the primary tank, being welded to the tank cover and reactor outlet piping. The casing contains a neutron shield which reduces the activation of the secondary sodium that flows through the tube bundle. The shield plug and tube bundle are an integral assembly that is removable at the operating floor. The tube bundle is unbaffled and is of conventional straight tube and tubesheet design, having 3,026 tubes of Type 304 stainless steel with 5/8-in. (15.9-mm) outside diameter X 0.052-in. (1.24-mm) wall thickness. Primary sodium flows downward on the shell side, and secondary sodium flows upward through the tubes. There are no provisions to compensate for differences in thermal expansion between tubes or the tubes and the center pipe. The IHX was installed in the EBR-II primary tank in November 1962. Operating conditions varied since 1963 as operating power was gradually increased to the design level of 62.5 MWT in September 1970. Since that time, operation has been at 62.5 MWT except for shutdown periods. By December 1977, the accumulated service will be approximately 3.5 million MWh. Except for a minor problem in November 1970, when a permanently installed evacuation tube came loose and was vibrating, service has been trouble-free. The evacuation tube was removed with the IHX in place with little difficulty and little loss of plant time.

- 652 (CONF-780251--P2, pp 8.199-8.230) CRBRP DECAY HEAT REMOVAL SYSTEMS. Hottel, R.E.; Louison, R.; Boardman, C.E.; Kiley, M.J. Aug 1978.

From US/USSR seminar on problems of design, development, fabrication, and test of breeder reactor components; Los Angeles, CA, USA (6 Feb 1978).

The Decay Heat Removal Systems for the Clinch River Breeder Reactor Plant (CRBRP) are designed to adequately remove sensible and decay heat from the reactor following normal shutdown, operational occurrences, and postulated accidents on both a short term and a long term basis. The Decay Heat Removal Systems are composed of the Main Heat Transport System, the Main Condenser and Feedwater System, the Steam Generator Auxiliary Heat Removal System (SGAHR), and the Direct Heat Removal Service (DHRS). The paper examines the overall design

of the CRBRP Decay Heat Removal Systems and the operation under normal and off-normal conditions and presents the redundancies of the system design, such as the four decay heat removal paths, the emergency diesel power supplies, and the auxiliary feedwater pumps; and the diversities of the design such as forced circulation/natural circulation and ac power/dc Power. In addition to overall design and system capabilities, the detailed designs for the Protected Air Cooled Condensers (PACC) and the Air Blast Heat Exchangers (ABHX) are presented.

- 653 (CONF-780251--P2, pp 10.289-10.314)

CRBRP SODIUM CIRCULATING PUMP DESIGN EVALUATION. Marrujo, F.; Cook, M.; Manners, L.; Cothran, H. Aug 1978.

From US/USSR seminar on problems of design, development, fabrication, and test of breeder reactor components; Los Angeles, CA, USA (6 Feb 1978).

One sodium pump is provided in the hot leg of each primary loop. Each of the pumps is nominally rated at 33,700 gpm (2.12 m³/sec), 458 ft (139.6m) total dynamic head. The pumps are driven by conventional squirrel-cage induction motors rated at 5,000 horsepower at 1,116 rpm. A pony motor rated at 75 horsepower is coupled to each pump motor by a reduction gear and overrunning clutch to drive the pump system at about 7-1/2 percent speed during standby operation, refueling, and decay heat removal periods. Each pump drive motor is powered by a commercial motor-generator set with fluid coupling. Drive speed turndown ratio is 4:5:1. The design of the sodium pumps is based on the free-surface centrifugal pump concept as successfully operated in EBR-II, EFBR, SRE, and HNPF. In this concept, the drive motor is located above the shield deck, the pump drive shaft penetrates the shield, and a buffer gas is contained between the sodium free surface and the final atmosphere seal at the pump shaft and containment boundary.

- 654 (CONF-780251--P2, pp 12.343-12.383)

ADVANCED WORK ON PUMPS AND HEAT EXCHANGERS. Jaross, R.; Anderson, R.; Brown, G.; Cox, J.; Brunsvold, A.; Nixon, D.; DeStefano, J. Aug 1978.

From US/USSR seminar on problems of design, development, fabrication, and test of breeder reactor components; Los Angeles, CA, USA (6 Feb 1978).

Model testing of an inducer for a centrifugal sodium pump plus test results is discussed. Hot and cold leg pump studies for liquid metal breeder reactor plants are summarized, including the operating parameters such as capacities from 77,000 gpm to 127,000 gpm, in addition to the key features of the conceptual designs and the needed supporting development work. A description of the ongoing program of the large sodium pump design for a primary hot leg and a secondary cold leg pump plus specification requirements and schedules is given. The feasibility study of a scale-up of the CRBRP IHX to commercial size, plus later low and high pressure drop commercial size conceptual studies is summarized. An advanced program for study of IHX maintainability including a tube plug placement device design, mock-up and testing, plus tube buckling analysis is outlined. A review of specific existing IHX designs, design criteria, and evaluation results is summarized.

- 655 (CONF-780251--P2, pp 11.315-11.342)

REVIEW OF FFTF AND CRBRP: CONTROL ROD SYSTEMS DESIGNS. Pitterle, T.A.; Lagally, H.D. Aug

1978.

From US/USSR seminar on problems of design, development, fabrication, and test of breeder reactor components; Los Angeles, CA, USA (6 Feb 1978).

Much of the design experience for the Clinch River Breeder Reactor (CRBR) design was developed in the design of the Fast Flux Test Facility (FFTF). CRBR designers were able to review the concepts developed for FFTF, integrate the current test data, and modify the designs and concepts as necessary to meet CRBR requirements. The primary control rod system design for CRBR progressed by the same evolutionary process, starting from the FFTF design concepts and integrating the considerable test data available and the specific requirements of CRBR. The paper describes the evolution of the primary control rod system design for FFTF and CRBR, beginning with the initial choice of the basic concepts. The significant component and systems tests are reviewed together with the test results which referenced the development of the CRBR primary control rod system design. Modifications to the concepts and detail designs of the FFTF control rod system were required principally to satisfy the requirements of CRBR, and at the same time incorporating design refinements shown desirable by the tests. The present status of the CRBR control rod system design is the inception of the systems and component test program to verify performance of the CRBR primary control rod system. The planned tests reflect the emphasis on reliability, and address the potential failure modes which the design experience has shown to be significant. It is expected that the experience from the current design experience and tests will contribute to future control rod systems designs as the FFTF experience contributed to CRBR.

656 (CONF-780251--P2, pp 6.107-6.139) FFTF AND CRBRP INTERMEDIATE HEAT EXCHANGER DESIGN, TESTING AND FABRICATION. Devlin, R.W.; Breshahan, J.D. Aug 1978.

From US/USSR seminar on problems of design, development, fabrication, and test of breeder reactor components; Los Angeles, CA, USA (6 Feb 1978).

The Heat Transport Systems of the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor Plant (CRBRP) include Intermediate Heat Exchangers (IHxs) to form barriers between the primary radioactive and secondary non-radioactive sodium coolant loops. The design, development and fabrication of the IHX for the FFTF has been completed and the units installed in the facility. The design and development of the IHX for the CRBRP has progressed through the performance sizing stage and flow testing of critical areas into early fabrication. The design evolution of these two heat exchangers may be considered a continuous development because the design and fabrication organizations have been the same for both units. Structural considerations to meet ASME Code requirements, support testing experience and manufacturing experience from the FFTF program has been factored into the CRBRP program. The paper outlines the similarities and differences between the FFTF and CRBRP designs. Special attention has been given to utilization of flow distribution devices and associated hydraulic testing, tube bundle and tube-to-tubesheet joint configurations, and fabrication differences.

657 (CONF-780251--P2, pp 7.141-7.197) FFTF DUMP HEAT EXCHANGER DESIGN AND DEVELOPMENT. Jansen, W.D.; Mahaffey, M.K.; Seber, E.C.;

Mancuso, R.A. Aug 1978.

From US/USSR seminar on problems of design, development, fabrication, and test of breeder reactor components; Los Angeles, CA, USA (6 Feb 1978).

The report is a brief summary of the history of procurement of the Dump Heat Exchangers for FFTF; it outlines the various organizations and respective roles. A description of the design basis and features of the large main heat transport system dump heat exchangers is included. Design evolution is discussed and testing performed as part of the design process, including finned tube qualification, header hydraulic tests, air-pressure drop tests, insulated panel tests and friction/wear testing, is summarized. Manufacturing experience, including development of an integral finned tube, development of welding procedures for joining the finned tubes, and header fabrication is also presented. Testing results obtained with the prototype unit are described, as are additional feature model tests that were performed to define possible methods of improving heat transfer performance for the plant units. The paper concludes with a summary of construction experience, status of plant modifications and a description of expected plant operation, including decay heat removal.

658 (CONF-780251--P2, pp 9.231-9.288) DESIGN EVOLUTION OF FFTF HEAT TRANSPORT SYSTEM PUMPS. Young, M.W.; Nixon, D.R.; Absher, K.R. Aug 1978.

From US/USSR seminar on problems of design, development, fabrication, and test of breeder reactor components; Los Angeles, CA, USA (6 Feb 1978).

The Fast Flux Test Facility (FFTF) Heat Transport System (HTS) pumps have been completed and are installed. The basic designs for the prototype, primary, and secondary pumps are described in the report. The descriptions include changes based on design reviews, high temperature and thermal transient considerations, sub-component development testing, and pump testing. Emphasis has been placed on changes resulting from pump tests in water and sodium test loops. Prototype and plant unit configurations are compared to show applicability of sodium test data to the plant units. The pump bearing failure experienced during thermal transient testing is also described, along with the changes required to prevent recurrence. Special prototype pump modifications were made to obtain data verifying adequacy of the design changes to the hydrostatic bearing. These changes are significant to future development programs, as are potential design improvements intended to reduce sensitivity to long-term dimensional instability and severe thermal transients.

659 (CONF-780374--1) ULTRASONIC INSPECTION FOR WASTAGE IN THE LMFBR STEAM GENERATOR DUE TO SODIUM-WATER REACTIONS. Neely, H.H.; Renger, L. (Atomics International Div., Richland, WA (USA). Rockwell Hanford Operations). 1977. Contract EY-76-C-04-3533. 10p. Dep. NTIS, PC A02/MF A01.

From Conference of the American Society for Nondestructive Testing; Phoenix, AZ, USA (28 Mar 1978).

As part of a program to study the results of large sodium-water reactions in the LMFBR Steam Generator, a boreside ultrasonic inspection device was developed to measure the wall thickness and diameter of the 2-1/4Cr-1 Mo, 0.397 in. I.D. steam tubes. The reaction was created in a near prototype steam generator by guillotine-type rupture of a steam tube, while

the generator was at operating conditions. Wastage occurred on the surrounding tubes due to the high temperature reaction. The UT test instrument was designed to operate with a 15 MHz transducer in the pulse-echo shear-wave mode, with a sampling rate of 10⁶/sec. System outputs are diameter, wall thickness, attitude and axial position of the transducer. All are displayed digitally and may be recorded. Measurements are fed into a computer for later retrieval, and/or cascaded outputs into an x-y recorded displaying either out-of-limit or thickness data. The UT data taken in this experiment were consistent with physical measurements on a tube which was removed from the generator after the test. A machined flat 1/8-inch long and 0.002-inch deep could readily be detected.

- 660 (CONF-780401-- , pp 191-206) USE OF INTEGRAL DATA IN THE DEVELOPMENT OF DESIGN METHODS FOR FAST REACTORS. Doncals, R.A.; Lake, J.A.; Paik, N.C. (Westinghouse Electric Corp., Madison, PA). Jun 1978.

From Topical meeting on advances in reactor physics; Gatlinburg, TN, USA (9 Apr 1978).

The paper describes an LMFB nuclear design methodology which has been strongly influenced by the availability of integral data, by the expansion of the differential nuclear data base, by improvements in large nuclear design computer codes, and by the specific reactor under consideration. The accuracy of the nuclear data base has been improved as the result of detailed differential measurements as well as extensive integral testing in the ZPR and ZPPR criticals. Due to the increased interest in radial parfait designs, the applicability of the design data and methods to the analysis of heterogeneous LMFB systems have been explored. The ability of the design data and methods to predict integral parameters in ZPPR is also discussed.

- 661 (CONF-780401-- , pp 479-492) PRESENT STATUS OF SODIUM VOID REACTIVITY PREDICTIONS IN CONVENTIONAL AND NONCONVENTIONAL FAST REACTOR CORE DESIGNS. Kuesters, H.; Ganesan, S. (Kernforschungszentrum, Karlsruhe, Ger.). Jun 1978.

From Topical meeting on advances in reactor physics; Gatlinburg, TN, USA (9 Apr 1978).

A review is given on the present status of the prediction of sodium void reactivity effects in conventional and nonconventional fast reactor core designs. First problems arising in the assessment of fast reactor safety are outlined and are discussed in relation to reactor physics aspects. Target accuracies for the sodium void effect are estimated. Nuclear data and calculational uncertainties are discussed, and the present accuracy achieved in various laboratories of the world is summarized. Some cases, for which the calculated void effects were rather far off the respective experimental results, were re-examined, resulting in significant improvements.

- 662 (CONF-780401-- , pp 517-525) CRITICALITY STUDIES ON PARTIALLY INSERTED ROD CONFIGURATIONS. Lee, B.W.; Indreboe, L.L.; Stewart, S.L. (General Electric Co., Sunnyvale, CA). Jun 1978.

From Topical meeting on advances in reactor physics; Gatlinburg, TN, USA (9 Apr 1978).

Criticality measurements were performed in Assembly 6 of ZPPR which included four core conditions. A bank of control rods was inserted various distances into the core for each configuration to simulate the first fuel cycle

of an LMFB. Analysis of criticality included both two-dimensional (X,Y) and three-dimensional (X,Y,Z) diffusion calculations. The core midplane compositions and the buckling values deduced from the axial leakage through the midplane were input to the (X,Y) calculation. The sensitivity of the axial leakage treatment in the (X,Y) calculations was analyzed. In addition, midplane neutron reaction rates were computed to provide diagnostic studies. It is concluded that the (X,Y) calculations using axial buckling deduced from the midplane axial leakage produce the best agreement with the 3-D calculations and provide a consistent prediction of criticality over a wide range of partially inserted configurations.

- 663 (CONF-780401-- , pp 527-538) NUCLEAR CHARACTERISTICS OF LARGE LMFB'S: HOMOGENEOUS AND HETEROGENEOUS DESIGNS. Lu, Y.S.; Cowan, C.L.; Bailey, H.S. (General Electric Co., Sunnyvale, CA). Jun 1978.

From Topical meeting on advances in reactor physics; Gatlinburg, TN, USA (9 Apr 1978).

GE-FBRD has carried out physics developments in the design of large size LMFBs using oxide fuel. A variety of designs have been investigated. Among those are the homogeneous (Pu,U)O₂ core, the heterogeneous (Pu,U)O₂ core, the homogeneous (²³³U,Th)O₂ core, and the heterogeneous (²³³U,Th)O₂ core. The nuclear characteristics of these cores are presented. The design data and core layout for each design are described. A discussion on the performance of these designs is given. It appears that the heterogeneous (Pu,U)O₂ core design possesses the potential of offering low sodium void worth advantage without significant economic impact and that the sodium voiding may not be an issue for the (²³³U,Th)O₂ cores.

- 664 (CONF-780401-- , pp 539-547) PERFORMANCE POTENTIAL OF (TH,U) CARBIDE AND (TH,U) NITRIDE FUEL IN 1200 MWE LMFB'S. Caspersson, S.A.; Kulwich, M.R. (Combustion Engineering, Inc., Windsor, CT). Jun 1978.

From Topical meeting on advances in reactor physics; Gatlinburg, TN, USA (9 Apr 1978).

An evaluation of the performance potential of thorium-uranium carbide and nitride fuel in 1200 MWE homogeneous and heterogeneous LMFB's has been completed. Comparisons were done with 9.40 mm outer diameter, 0.38 mm cladding fuel pins at a selected peak (3 sigma) linear power of 98.4 kW/m for all cores. Breeding ratios for the homogeneous (Th,U)C and (Th,U)N cores are 1.12 and 1.16, respectively. In the heterogeneous cores the breeding ratios are 0.08 to 0.09 greater. A comparative breeding ratio of 1.48 results for a sodium-bonded (U,Pu)C homogeneous LMFB with similar design criteria. Compound inventory doubling times for the homogeneous and heterogeneous (Th,U)C cores are 82 and 53 years, respectively. The enriched nitride cores, with a higher fuel density, show a large improvement in doubling time for the two core concepts, 53 and 46 years. The (U,Pu)C homogeneous core has a 10 year doubling time. Additional performance parameters have also been compared, including sodium void worths, specific inventories, fissile material gains, and equilibrium cycle reactivity losses.

- 665 (CONF-780401-- , pp 549-563) ALTERNATIVE FUEL CYCLES FOR BREEDER REACTORS. Bartine, D.E. (Oak Ridge National Lab., TN); Burns, T.J.; White, J.R.; Kallfelz, J.M.; Rowland, D.M. Jun 1978.

From Topical meeting on advances in reactor physics; Gatlinburg, TN, USA (9 Apr 1978).

The primary rationale for considering alternative fuel cycles is two-fold: possible reduced nuclear proliferation risk relative to the reference Pu/U cycle and the projected capital cost differential between fast and thermal reactor systems. One proposed alternate nuclear system has most reactors operating on a "denatured" fuel cycle, with fresh fuel containing ^{233}U mixed with sufficient ^{238}U so that it is unsuitable for weapons use; these reactors also contain fertile ^{232}Th for ^{233}U breeding. Any "plutonium burner" reactors would be inside a controlled safeguards area, and their primary goal would be to produce ^{233}U . The performance characteristics of various LMFBRs (both "classical" and "heterogeneous" configurations) are examined for each of the above roles. Calculations for denatured fuel show that in a heterogeneous design, which has mixed annuli of fertile and fissile elements, the ratio of ^{233}U production to plutonium production can be considerably increased over that for a classical LMFBR. The calculations have been performed primarily with the CITATION code, using ENDF/B-IV cross section data. For the designs considered, various significant design and performance parameters, such as the fissile production per cycle, are reported. Symbiotic system characteristics of various thermal/fast and fast/fast reactor combinations are discussed.

666 (CONF-780401-- pp 505-515) PURDUE UNIVERSITY FAST BREEDER BLANKET FACILITY. Ott, K.O.; Clikeman, F.M.; Fulford, P.J.; Borg, R.C.; Johnson, R.H.; Gailar, O.H. (Purdue Univ., Lafayette, IN). Jun 1978.

From Topical meeting on advances in reactor physics; Gatlinburg, TN, USA (9 Apr 1978).

The Purdue University fast breeder blanket facility has recently been completed. Neutrons for this subcritical facility are provided by a large ^{252}Cf source; cylindrical geometry is used. Transformer regions surround the source and are designed to match the spectrum of neutrons leaking from these regions with the spectrum of neutrons leaking from the core of a commercial fast breeder reactor. A radial blanket region surrounds the transformer regions; blanket loadings can be varied allowing a number of mock-ups of the radial blankets of proposed fast reactors. A variety of reaction rates and neutron energy spectra will be measured as a function of position in the blanket. These measurements will be compared with calculated spectra and reaction rates obtained using the same computational methods and cross sections which are currently being used to analyze proposed fast reactors. Measurements of gamma-ray heating rates in the blanket will be compared with transport calculations. Experiments using the Purdue facility will serve as integral tests of fast reactor blanket design techniques.

667 (CONF-780401--11) ASSESSMENT OF NUCLEAR DATA FILES VIA BENCHMARK CALCULATIONS: A REPORT ON THE NEACRP/IAEA INTERNATIONAL COMPARISON CALCULATION OF A LARGE LMFBR. LeSage, L.G.; Wade, D.C.; McKnight, R.D.; Freese, K.E. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 19p. Dep. NTIS, MF A01.

From Topical meeting on advances in reactor physics; Gatlinburg, TN, USA (9 Apr 1978).

Portions of document are illegible.

The results of an international comparison calculation of a large (1250 MWe) LMFBR benchmark model are presented and discussed. Eight reactor configurations were calculated. Parameters included with the comparison were: eigenvalue, $k_{\text{sub infinity}}$, neutron balance

data, breeding reaction rate ratios, reactivity worths, central control rod worth, regional sodium void reactivity, core Doppler and effective delayed neutron fraction. Ten countries participated in the comparison, and fourteen solutions were contributed. The discussion focuses on the variation in parameter values, the degree of consistency among the various parameters and solutions, and the identification of unexpected results. The results are displayed and discussed both by individual participant and by groupings of participants (e.g., results from adjusted data sets versus non-adjusted data sets). Unexpected large variations among results were observed for radial reaction rate and worth distributions and for the central control rod worth.

668 (CONF-780401--12) CONCLUSIONS DRAWN FROM SUBCRITICAL MULTIPLICATION RESULTS IN ZPPR. Carpenter, S.G.; McFarlane, H.F.; Lineberry, M.J.; Beck, C.L. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 13p. Dep. NTIS, PC A02/MF A01.

From Topical meeting on advances in reactor physics; Gatlinburg, TN, USA (9 Apr 1978).

Modified source multiplication (MSM) has been investigated as a measurement technique for several LMFBR designs in the 1000 MWt class. This investigation has been based on the results of hundreds of measurements in ZPPR as well as on detailed analysis of some of the components of the method. The results of this study have special significance for critical-experiment planning and for the use of ex-vessel detectors in operating LMFBRs. The calculated factors that are applied in the MSM technique for ex-core detectors were found to be particularly insensitive to the calculational method. This apparently resulted from two facts: (1) the calculated factors are ratios of ratios, which reduce the sensitivity to first order corrections, and (2) most neutrons which cause reactions in an ex-core detector originate in the small volume of core nearest to the detector. Maps of key parameters in the MSM technique have been generated to promote a general understanding of the method and to assure its proper application.

669 (CONF-780401--13) PHYSICS ASPECTS IN THE DESIGN OF HETEROGENEOUS CORES. Barthold, W.P.; Beitel, J.C. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 15p. Dep. NTIS (US Sales Only), PC A02/MF A01.

From Topical meeting on advances in reactor physics; Gatlinburg, TN, USA (9 Apr 1978).

In unprotected loss-of-flow transients in large LMFBRs the reactor may become super-prompt critical on sodium voiding alone. To preclude an energetic disassembly the subsequent fuel motion must be monotonically dispersive. It is difficult if not impossible to resolve details of fuel motion under super-prompt critical conditions at very high power levels. However, if the sodium void reactivity is reduced to zero or near zero values, the sensitivity of the energetics to details of fuel motion can be greatly reduced and even under pessimistic accident scenarios less energetic termination of the transient is expected. Therefore, there is a strong incentive for designing reactors that have a near zero value of sodium void reactivity. Among such designs, the heterogeneous configurations that consist of successive radial core and blanket zones are most promising. The enhancement of neutron leakage from the core zones into the internal blanket zones results in a low sodium void reactivity

because of the increase in the leakage component and decrease in the spectral component of the void reactivity.

Results for conventional and heterogeneous LMFBR designs are compared in some detail.

- 670 (CONF-780401--15) NUCLEAR DATA SENSITIVITY COEFFICIENTS FOR A (^{233}U - ^{232}Th) FUELED LMFBR. Turski, R.B.; McKnight, R.D. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 13p. Dep. NTIS, PC A02/MF A01.

From Topical meeting on advances in reactor physics; Gatlinburg, TN, USA (9 Apr 1978).

Sensitivity coefficients for the nuclear data of the principal constituents of a representative ^{233}U - ^{232}Th fueled LMFBR have been generated. Generalized perturbation coefficients for several reactor parameters were computed with the VARI code and ENDF/B Version 4 data for a homogeneous one-dimensional spherical model. The following ten reactor parameters were studied: eigenvalue; breeding ratio; inner core conversion ratio; outer core conversion ratio; central reaction rate ratios of f^{02} , c^{23} , and c^{02} relative to f^{23} ; and the central material reactivity worths of ^{233}U , ^{232}Th , and ^{23}Na . Total (energy-integrated) and energy-dependent sensitivity coefficients were obtained for each of these parameters to each of the following nuclear data: sigma/sub f/ for ^{233}U , ^{234}U , ^{232}Th , and ^{233}Pa ; sigma/sub c/ for ^{233}U , ^{234}U , ^{232}Th , ^{233}Pa , ^{23}Na , SS316 , and fission products; anti ν for ^{233}U and ^{232}Th ; and sigma/sub inel/ for ^{232}Th . These data are combined with estimated data uncertainties and with observed large differences in recent data evaluations to project the impact on integral parameters calculated for a ^{233}U - ^{232}Th fueled LMFBR.

- 671 (CONF-780401--16) UNCERTAINTIES IN THE BREEDING RATIO OF A LARGE LMFBR. Marable, J.H.; Weisbin, C.R. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 13p. Dep. NTIS, PC A02/MF A01.

From Topical meeting on advances in reactor physics; Gatlinburg, TN, USA (9 Apr 1978).

An uncertainty analysis for eight fast integral benchmark experiments was performed to find the most consistent set of values for both the integral and differential data as well as their associated covariances. Extensive covariance data including inelastic scattering for ^{238}U and uncertainties in the fission spectrum temperature were used in the adjustment. The resulting covariance for the adjusted data was folded with calculated performance parameter sensitivities for a two-dimensional model of a conventional homogeneous plutonium-oxide fueled fast breeder reactor. Based on the estimated covariance files it was found that the inclusion of integral experiment data reduced the uncertainty in the static breeding ratio to the design goal of two percent. Some apparent inconsistencies in the data were alleviated by including in an approximate way the effect of methods, uncertainties, and other possible systematic errors.

- 672 (CONF-780401--20) COMPARISON BETWEEN PHYSICS PARAMETERS IN CONVENTIONAL AND HETEROGENEOUS LMFBRs USING RESULTS FROM ZPPR. Collins, P.J.; Beck, C.L.; McFarlane, H.F.; Lineberry, M.J.; Carpenter, S.G. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 12p. Dep. NTIS, PC A12/MF A01.

From Topical meeting on advances in reactor physics; Gatlinburg, TN, USA (9 Apr 1978).

The analysis of several ZPPR experiments has been made using consistent methods and data.

- 673 (CONF-780407--6) FAST REACTOR OPERATION IN THE UNITED STATES. Smith, R.R.; Cissel, D.W. (Argonne National Lab., Idaho Falls, ID (USA)). 1978. Contract W-31-109-ENG-38. 33p. Dep. NTIS, PC A03/MF A01.

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

Of the many American facilities dedicated to fast reactor technology, six qualify as liquid-metal-cooled fast reactors. All of these satisfy the following criteria: an unmoderated neutron spectrum, highly enriched fuel material, substantial heat production, and the use of a liquid metal coolant. These include the following: EBR-I Clementine, LAMPRE, EBR-II, EFFBR, and SEFOR. Collectively, these facilities encompassed all of the more important features of liquid-metal-cooled fast reactor technology. Coolant types ranged from mercury in Clementine, to NaK in EBR-I, and sodium in the others. Fuels included enriched-uranium metallic alloys in EBR-I, EBR-II, and EFFBR; metallic plutonium in Clementine; molten plutonium alloy in LAMPRE; and a mixed UO_2 - PuO_2 ceramic in SEFOR. Heat removal techniques ranged from air-blast cooling in LAMPRE and SEFOR; steam-electrical generation in EBR-I, EBR-II, and EFFBR; to a mercury-to-water heat dump in Clementine. Operational experience with such diverse systems has contributed heavily to the U.S. Each of the six systems is described from the viewpoints of purpose, history, design, and operation. Attempts are made to limit descriptive material to the most important features and to refer the reader to a few select references if additional information is needed.

- 674 (CONF-780407--7) EBR-II COVER-GAS CLEANUP SYSTEM. Monson, L.R.; McDaniel, M.J.; Knox, A.E.; Rice, R.E. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 16p. Dep. NTIS (US Sales Only), PC A02/MF A01.

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

Operation of EBR-II with breached elements results in the continuous release of fission gases to the argon cover gas. To control activity in the reactor, a cover-gas cleanup system (CGCS) was installed to remove xenon and krypton from the cover gas by cryogenic distillation. Although only one breached fuel element will, by intention, be in-core during the early stages of the run-beyond-cladding-breach (RBCB) program now under way, the CGCS is designed to handle the activity released by up to 12 mixed-oxide elements.

- 675 (CONF-780440--1) ROLE AND PROGRESS OF THE CLINCH RIVER BREEDER REACTOR PLANT. Nolan, J.E.; Goesser, D.K. (Westinghouse Electric Corp., Oak Ridge, TN (USA). Advanced Reactors Div.). 1978. Contract EY-76-C-15-2395; EY-76-C-15-0003. 12p. Dep. NTIS, PC A02/MF A01.

From American power conference; Chicago, IL, USA (Apr 1978).

The Clinch River Breeder Reactor Plant plays a vital role in the development of an LMFBR industry by demonstrating the technical performance, reliability, maintainability, safety, environmental acceptability and economic feasibility of an LMFBR central

station power plant in a utility environment and confirms the value of the LMFBR in conserving natural resources. Activities already completed on the CRBRP Project have supported the attainment of these objectives in fulfilling the role of the CRBRP. In particular, the licensing interactions with the NRC leading to the issuance of a favorable or environmental acceptability (FES), the development of an industrial base of qualified component suppliers, and the completion of system and component designs, have all shown the ability of the CRBRP to meet its objectives. The most striking achievements on the CRBRP to date have been the completion of component designs and the considerable progress in component fabrication. Many of the major components are 50% or more through the fabrication process. The capability of the Project to maintain cost and schedule in these component procurements has been a major achievement and provides a solid basis for establishing the commercial viability of the LMFBR.

a systematic and disciplined evaluation of the plant design to search for and identify potential accident scenarios and to provide additional early assurance that important safety considerations have been identified. Results of the study indicate that CRBRP operational risks are small when compared with other societal risks and are comparable to those associated with current-generation Light Water Reactors (LWRs).

- 676 (CONF-780469--1) SOME RESULTS FROM THE TESTING OF A HIGH TEMPERATURE FISSION COUNTER-CHAMBER IN THE EBR-II J-2 THIMBLE OF THE NUCLEAR INSTRUMENT TEST FACILITIES (NITF). Yingling, G.E.; Taboas, A.L.; Popper, G.F.; Lipinski, W.C. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 14p. Dep. NTIS, PC A02/MF A01.

From IAEA symposium on nuclear power plant control and instrumentation; Cannes, France (24 Apr 1978).

A summary of test results from the life testing of a Twentieth Century Electronics (TCE) Model DC-14B fission counter-chamber at temperatures up to approximately 560°C in a typical Liquid Metal Fast Breeder Reactor (LMFBR) nuclear environment. The test results, after approximately 2.5 years of testing in the Experimental Breeder Reactor-II (EBR-II) J-2 thimble of the Nuclear Instrument Test Facilities (NITF), indicate that this type of detector, when attached to a current sensitive type preamplifier can be successfully utilized for long term, fixed position, low level and/or wide range neutron flux monitoring applications in thermal equivalent neutron fluxes of greater than 10^{11} nv. Brief descriptions of the test program, electronic test instruments used and the reactor facility are also included.

- 677 (CONF-780507--15) CLINCH RIVER BREEDER REACTOR PLANT SAFETY STUDY: SUMMARY AND RESULTS. Wood, P.J.; Leaver, D.W.W.; Piper, H.B.; Buhl, A.R.; Crump, R.J.; Hale, J.P.; Ferguson, D.R.; Griffin, C.W. (Clinch River Breeder Reactor Plant Project Office, Oak Ridge, TN (USA)). 1978. Contract EY-76-C-15-1200. 20p. Dep. NTIS, PC A02/MF A01.

From Probabilistic analysis of nuclear reactor safety; Los Angeles, CA, USA (8 May 1978).

The Clinch River Breeder Reactor Plant (CRBRP) Safety Study is an assessment of risks to the public associated with the operation of a 380-MW(e) liquid-metal fast breeder reactor (LMFBR) of the type to be built on the Clinch River site in Oak Ridge, Tennessee. The objectives of the study were to provide a realistic assessment of accident risks associated with the CRBRP, to place them in perspective in relation to other societal risks, and to aid in determining whether accident risks from the CRBRP are comparable to those for previously licensed reactors. Although this study was necessarily not completed with the depth or precision possible for operating reactors, its timing has allowed

- 678 (CONF-780622--15) VIBRATION STUDIES OF PERFORATED PLATES ON CRBRP MIXING TEE MODEL. Chung, H.; Halle, H. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 5p. Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

Results are reported for a series of vibration tests on a quarter scale model of a mixing tee for the CRBR intermediate heat transport system. The tests consisted of free vibration tests and flow tests on the perforated plates in the tee.

- 679 (CONF-780622--37) SIMPLE TRANSIENT ANALYSIS OF THE THORS BUNDLE 6A BOILING TEST RESULTS. Garrison, P.W. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 2p. Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

A simple one-dimensional boiling model has been developed to investigate the effect of bundle insulation degradation on the results of boiling stability tests conducted in the Thermal Hydraulic Out-of-Reactor Safety (THORS) Facility at ORNL. Significant periods (20 to 90s) of stable boiling without dryout were observed during a series of flow coastdown initiated boiling tests of a 19-pin simulated LMFBR fuel subassembly at power levels of 1.0 to 5.2 kW/ft.

- 680 (CONF-780622--42) SAS3A SIMULATIONS FOR A 19-PIN DRYOUT EXPERIMENT. Han, J.T.; Tomlinson, E.T. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 6p. Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

Three preliminary calculations using the one-dimensional SAS3A code have been performed to simulate a dryout experiment obtained in a 19-pin sodium-cooled bundle. Different values of the structure thermal inertia were used in SAS to account for the boiling incoherency in the test caused mainly by the wall heat loss. It was found that the larger the value used in SAS, the further the delay of boiling and dryout inception. In order to predict these events closely, two SAS calculations were required using different structure thermal inertias.

- 681 (CONF-780622--43) SENSITIVITY ANALYSES OF FAST REACTOR SYSTEMS INCLUDING THORIUM AND URANIUM. Marable, J.H.; Weisbin, C.R. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 4p. Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

The Cross Section Evaluation Working Group (CSEWG) has, in conjunction with the development of the fifth version of ENDF/B, assembled new evaluations for ^{232}Th and ^{233}U . It is the purpose of this paper to describe briefly some of the more important features of these evaluations relative to ENDF/B-4 to

project the change in reactor performance based upon the newer evaluated files and sensitivity coefficients for interesting design problems, and to indicate preliminary results from ongoing uncertainty analyses.

682 (CONF-780622--67) DESIGN APPLICATIONS OF IRRADIATION CREEP AND SWELLING DATA.

Boltax, A.; Foster, J.P.; Kalinowski, J.E.; Swenson, D.V. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1978. Contract EY-76-C-02-3045-007. 20p. Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

Recently, high dose irradiation creep and swelling data have been reported for a variety of stainless steels. However, the data base is still not sufficiently large for any one material to permit unambiguous determination of the functional dependencies of irradiation creep and swelling on dose, temperature, and stress. The purpose of the paper is to present an assessment of irradiation creep and swelling relative to various design applications and, in particular, to illustrate the sensitivity of uncertainties in the available data when applied to design calculations where irradiation creep and swelling act in opposition.

683 (CONF-780819--13) AGGLOMERATION CHARACTERISTICS OF FAST REACTOR HCDA AEROSOLS.

Parker, G.W.; Creek, G.E.; Sutton, A.L. Jr. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 26p. Dep. NTIS, PC A03/MF A01.

From 15. nuclear air cleaning conference; Boston, MA, USA (Aug 1978).

The behavior of vaporized mixed oxide fuel aerosols postulated to result from fast reactor core disruptive accidents is a subject which is intensely evaluated in fast reactor safety analysis, containment design, and site selection licensing procedure. In this program, surrogate uranium oxide aerosols produced by vapor condensation of super-heated liquid UO_2 have been produced in a variety of ways and have been intercompared with respect to particle morphology, number concentration, deposition behavior, surface area, oxygen content, and diffusion, sedimentation, and agglomeration rates. It has been established that, within a relatively narrow range, the primary particles are nearly identical when examined for initial shape, and size distribution, and effective density. However, primary particles in those aerosols produced by the most rapid heating methods followed by nearly instantaneous cooling are smaller in size and generally spherical, while those produced by slower processes in combination with delayed cooling give rise to a large proportion of crystalline particles which result in more tightly bound agglomerates. Deposition behavior is not significantly affected by this difference. Stokes diameters, which can be rather accurately calculated from the rate of settling, compare favorably with aerodynamic mean diameters measured with inertial devices. Agglomerate sizes appear to follow the cube root of the maximum initial concentration. The HAARM-2 version of the HAA-3 aerosol code has been used to obtain detailed time-dependent predicted behavior on a reference closed volume. The interrelations observed suggest that, in a given system, many of the characteristics can be calculated after only a few cases have been experimentally measured.

684 (CONF-780819--P1, pp 195-207) STUDY ON THE TRITIUM REMOVAL FROM THE SODIUM IN LMFB. Hata, K.; Nishizawa, Y.; Osawa, Y. (Mitsubishi Atomic Power Industries, Inc., Omiya, Japan). Feb 1979.

From 15. nuclear air cleaning conference; Boston, MA, USA (Aug 1978).

Removal of tritium in the sodium coolant of LMFB, is important for achieving the release of radioactivities as low as reasonably achievable. It is well known that cold trap should be an effective method for controlling tritium in sodium. To develop an effective tritium trapping system with reactive getter (chemical trap), in-sodium hydrogen absorption behavior has been studied. The absorption experiment was conducted by dipping some test materials into sodium on which cover gas was composed of an $Ar-H_2$ mixture. Hydrogen analysis and metallographic examination for the test materials were made after the experiment. From these experiments, it was concluded that Yttrium canned with Nickel or Niobium was the most suitable as reactive getter. Further investigation to develop the chemical trap made from these materials for practical application has been in progress.

685 (CONF-780819--P1, pp 257-265) STUDY OF THE CONTAINMENT SYSTEM OF THE PLANNED SNR-2 FAST BREEDER REACTOR. Bunz, H. (Kernforschungszentrum Karlsruhe GmbH, Germany); Scholle, U. Feb 1979.

From 15. nuclear air cleaning conference; Boston, MA, USA (Aug 1978).

Research is presented concerning the behaviour of four possible containment concepts for the planned Fast Breeder Reactor SNR-2 under the conditions of a HCDA. Detailed descriptions and calculations are given dealing with the acceptable leakages and the filter loads reached. The transport of the aerosol-type as well as the gaseous activity through the different compartments into the environment is calculated and from this the resulting accident doses. To cover the uncertainties of the accident course a wide range of accident parameters is examined.

686 (CONF-780819--P2, pp 764-791) SODIUM OXIDE/HYDROXIDE AEROSOL PROPERTIES AND BEHAVIOR IN A LARGE VESSEL. Hilliard, R.K.; McCormack, J.D.; Hassberger, J.A.; Postma, K.A. (Westinghouse Hanford Co., Richland, WA). Feb 1979.

From 15. nuclear air cleaning conference; Boston, MA, USA (Aug 1978).

Two large-scale aerosol behavior tests were performed in the Containment Systems Test Facility (CSTF) using sodium pool fires as the aerosol source. The purpose was to characterize the properties and behavior of sodium aerosol particles formed and aged in a large containment vessel. The results will be used as baseline data for comparison with follow-on air cleaning tests. The 20-m high, 850-m³ CSTF containment vessel is approximately half scale of a commercial reactor containment building for parameters which affect agglomeration and gravitational settling. Thus, the test air cleaning systems will be challenged with an aerosol closely representative of that postulated for LMFB accidents. The results of these tests also offer an excellent opportunity to evaluate computer models which have been developed for the purpose of predicting aerosol behavior during hypothetical accidents in sodium cooled reactor containment spaces. The two tests differed only in that steam was injected in the second test to simulate the release of water vapor from heated concrete. The aerosol in the first test was a dry mixture

of sodium peroxide, sodium hydroxide and sodium carbonate. In the second test it was predominantly wet sodium hydroxide. The addition of water vapor caused more rapid fallout during the aerosol release period, but lower settling rates afterward. The net results was a 25% lower potential for aerosol leakage in the wet test. Detailed analyses of computer model comparisons have not yet been performed, but preliminary HAA-3B code predictions were in good agreement with suspended mass concentration and the ratio of mass settled to plated on walls.

- 687 (CONF-780819--P2, pp 792-804)
ATTENUATION OF PARTICLES THROUGH LEAK PATHS.
Nelson, C.; Johnson, R. (Atomics International Div., Canoga Park, CA). Feb 1979.

From 15. nuclear air cleaning conference; Boston, MA, USA (Aug 1978).

Particulates of sodium oxide were passed through narrow leak passages. The particulates impacted on the walls and built up plugs which eventually stopped the flow. Conditions under which plugging occurs were determined. Material which did pass through the leaks was increased in size because of agglomeration on the wall followed by detachment. The data shows that radioactive particulates escaping through leaks following a hypothetical FBR accident will tend to plug the leaks in both the reactor vessel and the containment building. Also, the particulates will generally increase in size so as to be nonrespirable.

- 688 (CONF-780819--P2, pp 1044-1057)
PERFORMANCE OF A WET CELL WASHER FOR SODIUM FIRE AEROSOLS. Hinds, W.; Price, J.; First, M.W. (Harvard Air Cleaning Lab., Boston, MA). Feb 1979.

From 15. nuclear air cleaning conference; Boston, MA, USA (Aug 1978).

The suitability of 6 packing materials for removal of sodium fire aerosols in a wet cell scrubber was evaluated experimentally by measuring the efficiency and flow resistance of two 10.2 cm deep cells in series at air face velocities of 185 and 289 cm/sec, water flow rates of 0.210 and 0.387 cm³/sec/cm², with and without aerosol prehumidification. The most satisfactory material was found to be 50 µm diameter stainless steel fibers at a packing density of 30 kg/m³ (porosity = 0.99). Two 10.2 cm deep cells in series gave a removal efficiency of 85% for typical sodium fire aerosols at a face velocity of 289 cm/sec. Measurement of efficiency as a function of particle size indicated a reduction in efficiency in the 0.4 to 1.1 µm size range for all packing materials.

- 689 (CONF-780922--2) MAINTENANCE OF CLINCH RIVER BREEDER REACTOR NUCLEAR ISLAND SYSTEMS. Ehrman, C.S.; Burgess, D.S. (Burns and Roe, Inc., Oradell, NJ (USA); Atomics International Div., Canoga Park, CA (USA)). 1978. Contract EN-77-C-03-1547. 18p. Dep. NTIS, PC A02/MF A01.

From ASME conference on power generation; Dallas, TX, USA (10 Sep 1978).

The Clinch River Breeder Reactor Plant is being designed to be maintained so that a high plant availability goal can be achieved. The plant is being designed so that equipment disassembly, cleaning, repair, reassembly, and return to service can be conducted in an expeditious manner. The layout of the plant and of individual cells provides adequate accessibility, work clearance, laydown areas, removal paths, and handling areas. Wherever

possible, the plant systems have been designed to provide capability for hands-on inspection and maintenance without undue hazard to maintenance personnel. Major sodium system components are removed by special transfer and bag-out techniques and placed in shielded casks when required. Systems and facilities have been provided for removal of sodium and decontamination of components, supported by Radwaste Systems which provide for control and disposition of radioactive wastes generated or released during maintenance.

- 690 (CONF-780963--1) EBR-II: SUMMARY OF OPERATING EXPERIENCE. Perry, W.H.; Leman, J.D.; Lentz, G.L.; Longus, K.J.; Olson, W.H.; Shields, J.A.; Wolz, G.C. (Argonne National Lab., Idaho Falls, ID (USA)). 1978. Contract W-31-109-ENG-38. 78p. Dep. NTIS, PC A05/MF A01.

From Seminar on plant engineering; Mito, Japan (20 Sep 1978).

Experimental Breeder Reactor II (EBR-II) is an unmoderated, sodium-cooled reactor with a design power of 62.5 Mwt. The primary cooling system is a submerged-pool type. The early operation of the reactor successfully demonstrated the feasibility of a sodium-cooled fast breeder reactor operating as an integrated reactor, power plant, and fuel-processing facility. In 1967, the role of EBR-II was reoriented from a demonstration plant to an irradiation facility. Many changes have been made and are continuing to be made to increase the usefulness of EBR-II for irradiation and safety tests. A review of EBR-II's operating history reveals a plant that has demonstrated high availability, stable and safe operating characteristics, and excellent performance of sodium components. Levels of radiation exposure to the operating and maintenance workers have been low; and fission-gas releases to the atmosphere have been minimal. Driver-fuel performance has been excellent. The repairability of radioactive sodium components has been successfully demonstrated a number of times. Recent highlights include installation and successful operation of (1) the hydrogen-meter leak detectors for the steam generators, (2) the cover-gas-cleanup system and (3) the cesium trap in the primary sodium. Irradiations now being conducted in EBR-II include the run-beyond-cladding breach fuel tests for mixed-oxide and carbide elements. Studies are in progress to determine EBR-II's capability for conducting important 'operational safety' tests. These tests would extend the need and usefulness of EBR-II into the 1980's.

- 691 (CONF-780963--2) TRAINING EXPERIENCE AT EXPERIMENTAL BREEDER REACTOR II. Driscoll, J.W.; McCormick, R.P.; McCreery, H.I. (Argonne National Lab., Idaho Falls, ID (USA)). 1978. Contract W-31-109-ENG-38. 72p. Dep. NTIS, PC A04/MF A01.

From Seminar on plant engineering; Mito, Japan (20 Sep 1978).

The EBR-II Training Group develops, maintains, and oversees training programs and activities associated with the EBR-II Project. The group originally spent all its time on EBR-II plant-operations training, but has gradually spread its work into other areas. These other areas of training now include mechanical maintenance, fuel manufacturing facility, instrumentation and control, fissile fuel handling, and emergency activities. This report describes each of the programs and gives a statistical breakdown of the time spent by the Training Group for each program. The major training programs for the EBR-II Project are presented by multimedia methods at a pace

controlled by the student. The Training Group has much experience in the use of audio-visual techniques and equipment, including video-tapes, 35 mm slides, Super 8 and 16 mm film, models, and filmstrips. The effectiveness of these techniques is evaluated in this report.

- 692 (CONF-781001--P1) THIRD US/USSR LMFBF STEAM GENERATOR SEMINAR. VOLUME 1. US PAPERS. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 1978. Contract EY-76-C-03-0893-010. 356p. AT.
From 3. US-USSR LMFBF steam generator seminar; Tampa, FL, USA (Oct 1978).
Separate abstracts were prepared for the individual papers presented at the meeting.

- 693 (CONF-781001--P1, pp 1.1-1.30) DESIGN, MANUFACTURING, AND TEST STATUS OF THE CRBRP HOCKEY STICK STEAM GENERATORS. Lillie, A.F.; Bashford, A.L. 1978.
From 3. US-USSR LMFBF steam generator seminar; Tampa, FL, USA (Oct 1978).
The history of the development of the hockey stick steam generator, currently being designed and fabricated for use in the CRBRP, is traced back to its inception in the late 1960's. Additional supporting development, which has formed the basis for the current CRBRP design, is presented. The design requirements for the CRBRP steam generators is outlined and the basic design, which has been evolved to meet those requirements, is described. Of particular interest to the designer are the design analysis problem areas which have been encountered and the method of attack which has been employed to resolve these problems. Finally, the current status of the prototype and CRBRP steam generator plant unit production effort is described. Tube-tubesheet welding and welding of shell sections is discussed.

- 694 (CONF-781001--P1, pp 2.1-2.12) FABRICATION AND INSPECTION OF TUBING, TUBESHEETS, AND TUBE-TO-TUBESHEET WELDS FOR THE CRBRP STEAM GENERATORS. Spalaris, C.N.; Durand, R.E.; McClung, R.W.; Wright, E.A. 1978.
From 3. US-USSR LMFBF steam generator seminar; Tampa, FL, USA (Oct 1978).
The rationale for the technical details leading to the selection of tubing and tubesheet specifications and inspection techniques was based upon the desire to optimize the chances of success for the CRBRP steam generators. Fabrication processes, especially welding for the tube-to-tubesheet joints were recognized as critical. Thus, heavy emphasis was placed upon the development of reliable welding techniques as well as inspection methods using the most accurate and sensitive tools available. Accomplishments to date include: (a) the procurement of the tubing and tubesheets, to the desired specifications; (b) validation that the design base properties of wrought products are well within code acceptable levels; and (c) development of welding and tube-to-tubesheet inspection methods to provide satisfactory levels of quality.

- 695 (CONF-781001--P1, pp 3.1-3.34) DESIGN APPROACHES AND PROGRAMS FOR DEVELOPMENT OF STEAM GENERATORS FOR U.S. LMFBF PLANTS BEYOND CRBRP. Stone, C.C. (Argonne National Lab., IL); Ford, J.A.; McDonald, J.S.; Lohmeier, A.; Grant, G. 1978.
From 3. US-USSR LMFBF steam generator seminar; Tampa, FL, USA (Oct 1978).
The advanced steam generator program in the

U.S. for LMFBF Plants consists of base technology and development of three large-scale steam generator design concepts. A feasibility scale-up of the CRBRP hockey stick design steam generator for application in the Prototype Large Breeder Reactor (PLBR) has been performed by the Atomic International Division of Rockwell International. A single-wall helical-coil steam generator is being developed by Babcock and Wilcox Company and a double-wall straight-tube concept is being developed by the Tampa Division of the Westinghouse Electric Corporation. This paper gives a description of each of the three concepts, basis for selection and the main factors considered. The development that has been completed in support of the hockey stick design is described and the planned development and testing programs to support the double-wall and helical-coil designs are outlined. Two alternate steam cycles are being considered and a 70 MWt prototype for both the double-wall and helical-coil concepts is planned to be designed, fabricated and tested after selection of a single reference steam cycle for each of the concepts.

- 696 (CONF-781001--P1, pp 4.1-4.12) DECARBURIZATION OF 2-1/4CR-1MO IN FLOWING SODIUM. Roy, P.; Hampton, L.V.; Spalaris, C.N. 1978.
From 3. US-USSR LMFBF steam generator seminar; Tampa, FL, USA (Oct 1978).
The Intermediate System Mock-up Sodium Loop (ISML) was designed to model a bi-metallic (austenitic stainless and 2-1/4Cr-1Mo ferritic steel) LMFBF intermediate heat transport system. The objective of the loop is to establish decarburization kinetics of 2-1/4Cr-1Mo as well as carburization rates for Types 304 and 316 stainless, operative in an intermediate heat transport sodium loop. It is intended to verify design basis assumptions made for the CRBRP regarding carbon transport in the secondary system. This loop has operated for about 11,000 hours. The specimens tested include prototypical materials used for the fabrication of CRBRP steam generators, as well as experimental heats controlled to test the effects of remelt refining processes. Activation energies have been obtained for decarburization rates for electroslag remelt (ESR) and vacuum arc remelt (VAR) materials produced from the same air melted ingot. Geometry effects of test samples have been investigated and time dependence was established for the decarburization constants. Carburization rates in this loop for Type 304 stainless were found to be lower than the previous data band.

- 697 (CONF-781001--P1, pp 5.1-5.32) CORROSION AND IMPURITIES DEPOSITION EFFECTS UNDER DNB CONDITIONS. Rush, D.J.; Hampton, L.V.; Spalaris, C.N.; Schmidt, D.K.; Padden, T.R. 1978.
From 3. US-USSR LMFBF steam generator seminar; Tampa, FL, USA (Oct 1978).
A summary of two separate steam generator model test programs, including a description of the test models, test systems, control and monitoring of water chemistry, in situ nondestructive examination methods and results, post-test destructive examination methods and results, are presented. The test programs described are: the DNB Corrosion Effects test program conducted by General Electric which comprised a sodium-heated single-wall-tube test model with a single-wall heat transfer tube prototypic of the CRBRP steam generator tubing; and the Small Steam Generator Model corrosion test program conducted by Westinghouse which

comprised a sodium-heated double-wall tube. The objectives of these tests were to determine whether accelerated or localized damage would occur as a result of DNB operation. The two parameters varied were caustic level and heat transfer. The GE, or prototypic, test was conducted at an extreme condition of the plant specifications: 9 ± 3 ppb Na and thermal conditions corresponding to 64% plant power where the heat flux and tube wall temperature oscillations are maximum. The Westinghouse test was conducted at a 30 ppb sodium ion content to simulate worst-case conditions under which the CRBR might be operated with the thermal conditions corresponding to the less severe 100% power condition.

- 698 (CONF-781001--P1, pp 6.1-6.22) TUBE/SPACE PLATE INTERACTIONS IN SODIUM COOLED FAST BREEDER STEAM GENERATORS. Wilson, W.L.; Galioto, R.A.; Ring, P.J.; Page, J.P. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

The friction, wear, and self-welding behavior of 2-1/4Cr-1Mo steam generator tubing interacting with tube space plates has been investigated under prototypic LMFBR steam generator service conditions. Dynamic friction and self-welding characteristics were investigated. Subsequent testing involved tube/tube space plate model tests utilizing actual steam generator tubing and spacer plate materials and configurations. Conditions of contact load, stroke, velocity, rubbing distance, sodium temperature, and intermittent dwell periods were investigated. In addition, the behavior of alternate spacer plate materials was evaluated. Dynamic coefficients of friction and breakaway friction coefficients were measured in sodium. Inconel 718 spacer plate material was found to result in a marked reduction in tube wear under all test conditions as compared to the reference 2-1/4Cr-1Mo and alternate spacer plate materials which were evaluated.

- 699 (CONF-781001--P1, pp 7.1-7.30) HEAT TRANSFER WITH HOCKEY-STICK STEAM GENERATOR. Moody, E.; Gabler, M.J. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

The hockey-stick modular design concept is a good answer to future needs for reliable, economic LMFBR steam generators. The concept was successfully demonstrated in the 30 MWt MSG test unit; scaled up versions are currently in fabrication for CRBRP usage, and further scaling has been accomplished for PLBR applications. Design and performance characteristics are presented for the three generations of hockey-stick steam generators. The key features of the design are presented based on extensive analytical effort backed up by extensive ancillary test data. The bases for and actual performance evaluations are presented with emphasis on the CRBRP design. The design effort on these units has resulted in the development of analytical techniques that are directly applicable to steam generators for any LMFBR application. In conclusion, the hockey-stick steam generator concept has been proven to perform both thermally and hydraulically as predicted. The heat transfer characteristics are well defined, and proven analytical techniques are available as are personnel experienced in their use.

- 700 (CONF-781001--P1, pp 8.1-8.28) THERMAL-HYDRAULICS IN A HELICAL COIL TUBE STEAM GENERATOR. Kakarala, C.R. 1978.

From 3. US-USSR LMFBR steam generator

seminar; Tampa, FL, USA (Oct 1978).

Thermal-hydraulic considerations play an important part in all stages of the design and development of a steam generator with a high degree of reliability and safety, ease of maintenance, and reasonable first cost. With this perspective, the subject of this paper is presented in three parts: (1) thermal-hydraulic considerations in the selection of the B and W helical coil tube once-through steam generator configuration and other important design parameters; (2) heat transfer and pressure drop correlations and the supporting experimental and theoretical basis; and (3) design and analysis methods for sizing, steady-state and dynamic performance evaluation. The present thermal-hydraulic design and analyses methods and back-up test data provide a sound basis for the design. Additional test programs, analytical model developments, building and testing a prototype, as described in a separate paper, will provide further assurance regarding the reliability of the design. These development programs will also allow further design optimization.

- 701 (CONF-781001--P1, pp 9.1-9.26) HEAT TRANSFER WITH DOUBLE WALL STEAM GENERATOR. Efferding, L.E.; Waszink, R.P.; Hwang, J.Y. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

Experimental/analytical studies performed to date on the double-wall tube heat transfer-hydraulic development have been addressed to single-tube sodium-heated experience and analysis support. These programs and results are summarized in this paper. Results demonstrate that thermal resistance behavior of the interface clearance of an annealed double-wall tube without grooves is predictable. References tubes employing grooves for leak detection reflect predictable interface thermal resistance behavior within 10% of experimental results derived from a sodium-heated test model. Flow boiling critical heat flux (CHF) data is shown to be unaffected by the behavior of the double-wall variable conductance interface. Use of a gas mixture in the interface of a double-wall tube test model demonstrated the versatility of this concept as a heat-flux-limiting feature without introducing a significant sizing penalty in other regions exposed to lower heat flux. Plans are described for future effort which applies these earlier findings to large steam generator tube bundles and describes the areas where future development is required. These areas of future effort entail thermal/hydraulic stability investigations of a large matrix of flow-boiling circuits, long-term test exposure of prototypic tubing at elevated-temperature, shell-side flow distribution of sodium through a baffled tube bundle reflecting two-dimensional (axisymmetric) and three-dimensional effects experiencing forced- and natural-circulation mechanisms.

- 702 (CONF-781001--P1, pp 10.1-10.20) RECENT EXPERIMENTAL RESULTS ON SMALL LEAK BEHAVIOR AND INTERPRETATION FOR LEAK DETECTION. Greene, D.A.; Gudahl, J.A.; Magee, P.M. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

A small leak of water into sodium in an LMFBR steam generator can cause damage to adjacent tubes, a phenomenon termed impingement wastage. The available impingement wastage data are analyzed by considering the sodium-water reaction to be a simple combustion process. This approach permits correlation of data from different facilities and integrates and

explains experimental observations. Water leaking into sodium also causes self-enlargement of the defect, or self-wastage. Results of recent experimental tests on the growth patterns of very small leaks are summarized and used to provide estimates of the magnitude of self-wastage phenomena. Steam generator leak protection is strongly influenced by the wastage results. Significant differences in protection exist between evaporators and superheaters due principally to their differences in sodium temperature. Two types of tube defect plugging have been studied: (1) plugging prior to steam generator service due to magnetic formation or the presence of foreign matter; and (2) plugging due to sodium-water reaction products. It is shown that a cleaning solution will open both types of plugs. Experimental studies on pool fire (diffusion flame) damage at sodium surfaces are briefly reviewed.

- 703 (CONF-781001--P1, pp 11.1-11.14) DEVELOPMENT OF LEAK DETECTION SYSTEMS FOR LMFBR STEAM GENERATORS IN THE U.S. Magee, P.M.; Gerrels, E.E.; Greene, D.A.; McKee, J. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

Leak detection for Liquid Metal Fast Breeder Reactor steam generators is an important economic factor in the shutdown, repair and restart of a plant. The paper summarizes U.S. progress in three key technology areas involved in leak detection development: (1) chemical (H_2 and O_2) leak detection meters; (2) acoustic leak detection/location techniques; and (3) data management for plant operations. The status, design aspects, and applications of leak detector technology for LMFBR plants are discussed.

- 704 (CONF-781001--P1, pp 12.1-12.30) U.S. PROGRAM FOR LARGE SODIUM/WATER REACTION TESTS. Whipple, J.C.; Amos, J.C.; DeBear, W.S.; Neely, H.H. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

An extensive large leak test program is underway in the United States to investigate the effects of large sodium-reaction events in LMFBR steam generators. The general objectives of the program are to provide data for validating analytical codes, provide information on the extent of secondary tube failures, provide data on relief system performance and provide data on cleanup and recovery needs. Tests are conducted in the Large Leak Test Rig located at the Energy Technology Engineering Center. The program is divided into two phases, Series I and Series II, for the purpose of satisfying near-term and long-term needs. The test article for the Series II tests (the Large Leak Test Vessel/Large Leak Test Internals) has been fabricated. Upon completion of Large Leak Test Rig modification, test article installation, and system checkout Series II testing will begin. The overall program plan is defined, the test articles and test facility are described, and the experimental approach is discussed for both Series I and Series II programs. The Series I test experience and results of intertest examinations are also presented.

- 705 (CONF-781001--P2) THIRD US/USSR LMFBR STEAM GENERATOR SEMINAR. VOLUME 2. USSR PAPERS. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 1978. 356p. AT.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

Separate abstracts were prepared for the individual papers presented at the meeting.

- 706 (CONF-781001--P2, pp 1.1-1.20) STATUS OF DEVELOPMENT AND MANUFACTURE OF A STEAM GENERATOR FOR THE BN-600 NUCLEAR POWER REACTORS. Titov, V.F.; Lukasevich, B.I. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

The basic results are presented from studies performed in the stage of development of the plan for a concurrent, multi-section steam generator for the BN-600 nuclear powerplant. Thermal and hydrodynamic, strength and vibration, heat-corrosion and operating-life testing of models of the steam generator have confirmed the plan assignments of specification parameters for the steam generator. One section of the PG-2 steam generator, the prototype of the actual steam generator of the BN-600, the manufacture of which has assisted in developing the technological cycle for production of the BN-600 steam generator, has been installed on the BOR-60 steam generator, has been installed on the BOR-60 experimental installation for long-term combined testing.

- 707 (CONF-781001--P2, pp 2.2-2.10) SOME ESTIMATES OF THE PROSPECTS FOR DUAL-WALL SODIUM-WATER STEAM GENERATORS. Kokorev, P.V.; Turchin, N.M. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

Steam generators with double tubes practically eliminate the probability of interaction of sodium and water in case of a leak. However, the opinion is still heard that these steam generators are significantly larger and heavier, and higher in initial cost than single-wall generators. Calculated estimates of the variation in heat-exchange surface as a function of thermal resistance of the zone of contact for steam generators with the parameters of the BN-600 installation have shown that the use of double tubes would not lead to an excessive increase in cost. Development of the design of double tubes, allowing easy repair of defective tubes, will make it desirable to manufacture steam generators of large unit power. These designs can be the equal of modular single-wall apparatus.

- 708 (CONF-781001--P2, pp 3.1-3.16) SOME RESULTS OF THE STUDY OF VIBRATIONS AND WEAR IN MODELS OF THE STEAM GENERATOR OF THE BN-600. Titov, V.F.; Fedorov, V.G. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

Results are discussed from testing models of the tubes of steam generators for fretting. The parameters of vibration in the models are selected from the results of measurement of vibrations of a multitube steam generator at nominal water flow rates. During the fretting tests, the amplitudes of tube acceleration, compressive forces of the tubes against the spacing grid and water flow rates were varied. A description is presented of the models, the testing methods and methods of processing the results. A comparison of the results produced with data produced earlier is performed and the results are used to estimate the operating life of steam generators.

- 709 (CONF-781001--P2, pp 4.1-4.10) RESULTS OF STUDY OF AMERICAN SPECIMENS OF WELDED JOINTS BETWEEN TUBES AND TUBESHEETS. Gubanov, V.M.; Sobolev, E.N.; Sokolov, A.S.; Titov, V.F.

1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

Results are presented from laboratory testing of specimens of welded joints between tubes and tubesheets provided by the American side. The tests were conducted in the USSR in accordance with the cooperative plan.

- 710 (CONF-781001--P2, pp 6.1-6.22) MODEL THERMAL CORROSION TESTING OF THE BN-600 STEAM GENERATOR. Titov, V.F.; Gubanov, V.M.; Sobolev, E.N.; Sokolov, A.S.; Lukasevich, B.I. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

Results are presented from thermal corrosion testing of models of the steam generator of the BN-600, operating life up to 12,000 hours, as well as information on the design of the models, the experimental test stand, the methods of conduct of the tests and of processing of the experimental results. Based on the experimental studies, it is shown that the 12KH2M steel used for the model evaporator and the KH18N9 steel used for the steam generator, under the planned operating conditions, can provide reliable operation of the steam generator for the assigned operating life.

- 711 (CONF-781001--P2, pp 9.1-9.22) PROBLEMS OF HEAT EXCHANGE AND HYDRODYNAMICS OF SODIUM-WATER STEAM GENERATORS FOR FAST-NEUTRON REACTORS. Kirillov, P.L.; Ushakov, P.A.; Grachev, N.S. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

This report contains a review of the problems and tasks related to the heat exchange and hydrodynamics of sodium-water steam generators, as related to the program of development of fast-neutron reactors. The hydrodynamics of these steam generators on the sodium side are significantly more important than has been thought up to now. Examples are presented of hydrodynamic effects. A general description of heat transfer from the sodium is discussed without reference to any specific design of steam generator. The characteristics of heat transfer on the water side and in the steam-water mixture are analyzed as applicable to the design of a steam generator with straight tubes. Information on heat-exchange crisis in a two-phase flow under various conditions, and the phenomena accompanying it are discussed separately, and problems of heat exchange in the supercritical zone of a steam generator are also discussed. A number of other phenomena are mentioned, related to deposits in steam generators, corrosion, plugging of tubes and other problems, which must be considered during design.

- 712 (CONF-781001--P2, pp 11.1-11.30) SMALL LEAKS OF WATER INTO SODIUM AND THEIR DETECTION BY CONCENTRATION METHODS. Kozlov, F.A.; Sergeyev, G.P.; Poplavskii, V.M.; Sednev, A.R.; Borisov, V.V.; Makarov, V.M. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

Reports are presented from studies of the spontaneous development of small leaks of water into the sodium, and methods of their recording on the basis of the hydrogen concentration in the sodium and in the protective gas are discussed. Based on a calculation model, an estimate is presented of the coefficient of hydrogen leakage into the gas space of a steam generator upon appearance of leaks of water into the sodium. The calculation results are

compared with the values of leakage coefficient determined experimentally. Possibilities of detecting leaks using hydrogen detectors are analyzed as applicable to the conditions of the BN-350 and BN-600 installations.

- 713 (CONF-781001--P2, pp 12.1-12.18) EXPERIMENTAL STUDY OF ABNORMAL PROCESSES WITH SMALL LEAKS OF WATER INTO SODIUM IN MODELS OF STEAM GENERATORS OF REVERSE DESIGN. Poplavskii, V.M.; Bagdasarov, Yu.E.; Kozlov, F.A.; Sergeyev, G.P.; Borisov, V.V.; Sednev, A.R. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

This report describes the basic peculiarities of reverse type steam generators, the advantages and disadvantages of the design. A description is presented of experimental devices used to study abnormal processes, and the research method is outlined. Based on the experimental results produced, conclusions are drawn concerning possible advantages of reverse steam generator design for sodium-water steam generators, from the stand-point of decreasing abnormal effects resulting from contact between the two media.

- 714 (CONF-781001--P2, pp 13.1-13.26) INFLUENCE OF HEAT EXCHANGE BETWEEN REACTION PRODUCTS OF THE INTERACTION OF SODIUM WITH WATER AND STRUCTURAL ELEMENTS OF THE STEAM GENERATOR ON THE PARAMETERS OF THE LARGE LEAK EMERGENCY MODE. Poplavskii, V.M.; Leonchuk, V.N. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

In this work, an attempt is made to analyze the temperature state of the material of the wall of a heat-transmitting tube in a steam generator in the area of a large leak of water into the sodium, considering its cooling by the steam-water flow, in order to predict the possibility of secondary ruptures. Physical and mathematical models of the process of heat exchange between the reaction products and the tube bundle of the steam generator in the large leak mode are suggested. The calculated results are compared with experimental results. The possibility of secondary ruptures is analyzed for the case of an instantaneous large leak of water into the sodium under the conditions found in a full-scale steam generator.

- 715 (CONF-781001--P2, pp 15.1-15.28) EXPERIENCE OF THE OPERATION OF LEAK-TESTING SYSTEMS OF SODIUM-WATER STEAM GENERATORS OF THE BN-350 INSTALLATION. Kozlov, F.A.; Kuznetsov, E.K.; Kozub, P.S.; Demin, A.A.; Samarkin, A.A.; Shvedenko, I.M.; Grebenshchikov, Yu.E. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

This report presents materials on the experience of operation of the leak detection system in the steam generators, used to test the hydrogen content in the protective gas and the sodium of the BN-350 installation. The selection of the system and its parameters are described, and examples are presented of the detection of leaks during operation of the steam generators. In the system which tests for leaks by hydrogen analysis in the protective gas, sodium vapor is trapped by a special filter. The use of a conductometric analyzer allows the detection of changes in hydrogen concentration in the protective gas of about 0.005 vol. %. It has been found that in some evaporators, the quantity of sodium contained in the flow of gas being analyzed is many times the calculated quantity. Results are presented from start-up and adjustment operations of the

system for testing of the hydrogen content of sodium, which uses a thick nickel membrane, as well as a test system with a thin nickel membrane in combination with a magnetic-discharge pump.

- 716 (CONF-781001--P2, pp 17.1-17.18) MAIN RESULTS OF EXPERIMENTS WITH A MAJOR LEAK OF WATER INTO SODIUM CONDUCTED IN NINETEEN-TUBE MODELS. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

The present report presents the main results of studies of effects that accompany major leaks of water into sodium under the conditions of 19-tube models in 1964--1967. The organization of the experiments and the methods of research are described. The dynamic and temperature phenomena in the sodium circuit and the vulnerability to damage of the tube bundle are discussed.

- 717 (CONF-781001--P2, pp 18.1-18.28) ACOUSTICAL METHODS OF LEAK DETECTION IN SODIUM-WATER STEAM GENERATORS (A SURVEY PAPER). Afanas'ev, V.A.; Sokolov, V.M. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

The results of preliminary studies aimed at determining the feasibility of leak detection in a sodium-water steam generator by the acoustical method are presented. Results of studies of noise generation aimed at substantiating the selection of the frequencies used in the leak detection device are discussed. A qualitative analysis is given of the mechanisms of noise generation in a gas jet, and some experimental findings are presented.

- 718 (CONF-781001--P2, pp 19.1-19.14) REPORT ON THE RESULTS OF ANALYSIS OF EMERGENCY LARGE LEAK MODES IN THE STEAM GENERATORS OF THE BN-350 INSTALLATION. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

This report briefly outlines basic results of analysis of emergency processes in the steam generator of the BN-350 reactor, resulting from large leaks of water into the sodium loop. A description is presented of the steam generator of the BN-350, and the characteristics of its emergency protection system are described.

- 719 (CONF-781001--P2, pp 20.1-20.6) RESEARCH PROGRAM ON AN ACOUSTICAL METHOD OF LEAK DETECTION IN SODIUM-WATER STEAM GENERATORS FOR 1978--1979. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

Bench tests on the characteristics of acoustic noise during a steam and gas leak into sodium are described. Tests using a model of the BN-600 steam generator are also described. Tests planned for the BOR-60 modular steam generator are mentioned.

- 720 (CONF-781001--P2, pp 5.1-5.22) STUDY OF THE ONCE-THROUGH SERPENTINE STEAM GENERATOR IN THE BOR-60 EXPERIMENTAL NUCLEAR POWERPLANT. Kondrat'ev, V.I.; Titov, V.F.; Bai, V.F. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

Results are presented from testing of a once-through serpentine steam generator which had operated under load for 18,000 hours in the BOR-60 installation. Its basic operational, steady-state, transient and emergency modes are

described, as well as the water chemistry determined in the process of testing. The results of corrosion-mechanical studies of the coils of the steam generator, made of 1Kh2M steel, are presented for testing after 4,000, 6,200, 8,400 and 18,000 hours of reactor operation. The nature, composition and thickness of corrosion products in the evaporated and steam-superheating sections are indicated. The degree of damage to the tubes of the coils resulting from pitting corrosion, decarburization and changes in the mechanical properties of the steel in the process of use is estimated.

- 721 (CONF-781001--P2, pp 14.1-14.18) SIMULATED AND EXPERIMENTAL STUDIES OF LARGE LEAKS IN MICROMODULAR STEAM GENERATORS. Chernobrovkin, Iu.V.; Kondrat'ev, V.I.; Smirnov, A.M.; Bukhteev, I.S.; Tomesh, V.; Matal, D. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

A program of experimental work is presented for the study of small and large leaks of water into the sodium in the micromodular steam generator of the BOR-60 reactor. Requirements are established as to the volume of information collected during experiments and the measurement systems required. The basic prerequisites for the method and programs for simulation of large-leak emergency are presented, with some results of leak simulation involving bursting of the entire cross section of one tube in the micromodular steam generator of the BOR-60.

- 722 (CONF-781001--P2, pp 7.1-7.18) STUDY OF DECARBURIZATION CONSTANTS OF PEARLITIC STEELS TYPE 2.25CR--1MO AND CARBURIZATION OF 12KH18N9 AND 304 STEELS. Starkov, O.V.; Kononov, E.S.; Kononyuk, M.Kh.; Umyashkin, E.V.; Kozlova, N.A. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

This report presents the results of investigation of the transfer of carbon from austenitic steels to pearlitic steels of Soviet and American manufacture. The tests were performed under isothermal conditions and in a nonisothermal stream of sodium in the 500 to 700°C temperature interval. Equilibrium concentrations of carbon in the steels 10Kh2M and 2.25CR--1Mo and decarburization constants were achieved. A study is made of the influence of carbon on the mechanical properties of pearlitic and austenitic steels, and this influence is evaluated for the specific usage conditions.

- 723 (CONF-781001--P2, pp 8.1-8.26) COMPARATIVE MATERIAL STUDIES OF SPECIMENS OF STEELS TYPE 2.25CR--1MO AND 18-8 OF SOVIET AND AMERICAN MANUFACTURE FOLLOWING CONTACT WITH SODIUM. Starkov, O.V.; Golubenkova, S.M.; Kondratenko, S.N.; Kononyuk, M.Kh.; Filin, M.P. 1978.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

This report presents the results of comparative studies of steels type 2.25Cr--1Mo and 18-8 of Soviet and American manufacture before and after testing in a nonisothermal sodium loop at $T = 510^\circ\text{C}$ for test times of up to 4550 hours and in a convection flow of sodium with a high content of oxygen and sodium hydroxide in the 450 to 730°C temperature interval.

- 724 (CONF-781001--P3) THIRD US/USSR LMFBR

STEAM GENERATOR SEMINAR. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Oct 1978. Contract EY-76-C-03-0893-010. 70p. AT.

From 3. US-USSR LMFBR steam generator seminar; Tampa, FL, USA (Oct 1978).

The development of liquid metal heated steam generators for use in an LMFBR is a major technological undertaking which can be assisted by international exchanges. The United States and Russian governments have negotiated an agreement which encourages technical interchanges of LMFBR data. The seminar held in Tampa in October was the 3rd such meeting to interchange information and views and discuss recent liquid metal steam generator technology developments. The document contains the Memorandum and Record of the seminar.

- 725 (CONF-781022--25) GENERIC CONSIDERATIONS OF LMFBR HYPOTHETICAL ACCIDENT ENERGETICS. Fauske, H.K.; Cho, D.H.; Epstein, M.; Grolmes, M.A.; Henry, R.E. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 14p. Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

The paper provides a preliminary assessment of generic accident energetics issues associated with alternatives relative to the reference (U,Pu) oxide fuel in liquid metal fast breeder reactors. The alternatives considered include thorium- and uranium-based oxide, carbide and metal fuel types. This assessment is made within the context of low probability, but potentially large consequence accidents, e.g., core-disruptive accidents.

- 726 (CONF-781022--27) STRUCTURAL AND CONTAINMENT RESPONSE TO LMFBR ACCIDENTS. Marchaterre, J.F.; Fistedis, S.H.; Baker, L. Jr.; Stepniewski, D.D.; Peak, R.D.; Glueckler, E.L. (Argonne National Lab., IL (USA); Hanford Engineering Development Lab., Richland, WA (USA); General Electric Co., Sunnyvale, CA (USA)). 1978. Contract W-31-109-ENG-38. 15p. Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

The adequacy of the containment of fast reactors has been traditionally evaluated by analyzing the response of the containment to a spectrum of core disruptive accidents. The current approach in the U.S. is to consider fast reactor response to accidents in terms of four lines of assurance (LOAs). Thus, LOA-1 is to prevent accidents, LOA-2 is to limit core damage, LOA-3 is to control accident progression and LOA-4 is to attenuate radiological consequences. Thus, the programs on the adequacy of containment response fall into LOA-3. Significant programs to evaluate the response of the containment to core disruptive accidents and, thereby, to assure control of accident progression are in progress. These include evaluating the mechanical response of the primary system to core disruptive accidents and evaluating the thermal response of the reactor structures to core melting, including the effects this causes on the secondary containment. The analysis of structural response employs calculated pressure-volume-time loading functions. The results of the analyses establish the response of the containment to the prescribed loadings. The analysis of thermal response requires an assessment of the distribution and state of the fuel, fission products and activated materials from accident initiation to final disposition in a stable configuration.

- 727 (CONF-781022--28) CRITICAL EXPERIMENTAL STUDY OF INTEGRAL PHYSICS PARAMETERS IN SIMULATED LMFBR MELTDOWN CORES. Bhattacharyya, S.K.; Wade, D.C.; Bucher, R.G.; Smith, D.M.; LeSage, L.G. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 10p. Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

A program of measurements designed to provide experimental data for the validation of analytical methods used in the neutronics part of LMFBR accident analyses has recently been completed at Argonne National Laboratory. A considerable volume of critical experimental data already exists for normal, undamaged LMFBR cores, and the size of the calculational and cross-section data errors is well established. However, this knowledge cannot be easily extrapolated to severely distorted accident configurations because of the large changes in neutron spectrum and system leakage properties (including streaming in large cavities) which are induced by redistribution of materials. The available experimental data on the physics of damaged LMFBR cores are inadequate for use in the systematic validation of calculational methods for a variety of reasons. The present program provides a clean, benchmark set of experiments with large reactivity changes (involving approximately 18.5% core volume distortions) for use in the methods validation. The reference configuration for the measurements consisted of a single core zone with a typical LMFBR composition. A sequence of idealized configurations was designed to follow from this reference to simulate, in a static or "snap-shot" sense, the sequence of configurations that might result from a loss-of-flow transient.

- 728 (CONF-781022--30) LMFBR OPERATIONAL SAFETY: THE EBR-II EXPERIENCE. Sackett, J.I.; Allen, N.L.; Dean, E.M.; Fryer, R.M.; Larson, H.A.; Lehto, W.K. (Argonne National Lab., Idaho Falls, ID (USA)). 1978. Contract W-31-109-ENG-38. 14p. Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

The mission of the Experimental Breeder Reactor II (EBR-II) has evolved from that of a small LMFBR demonstration plant to a major irradiation-test facility. Because of that evolution, many operational-safety issues have been encountered. The paper describes the EBR-II operational-safety experience in four areas: protection-system design, safety-document preparation, tests of off-normal reactor conditions, and tests of elements with breached cladding.

- 729 (CONF-781022--31) ANALYSIS OF HYPOTHETICAL LMFBR WHOLE-CORE ACCIDENTS IN THE USA. Ferguson, D.R.; Dettrich, L.W.; Brown, N.W.; Walter, A.E. (Argonne National Lab., IL (USA); General Electric Co., Sunnyvale, CA (USA); Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract W-31-109-ENG-38. 14p. Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

The issue of hypothetical whole-core accidents continues to play a significant role in assessment of the potential risk to the public associated with LMFBR operation in the USA. The paper briefly characterizes the changing nature of this role, with emphasis on the current risk-oriented perspective. It then describes the models and codes used for accident analysis in the USA which have been developed under DOE sponsorship and summarizes

some specific applications of the codes to the current generation of fast reactors. An assessment of future trends in this area concludes the paper.

- 730 (CONF-781022--32) LOW SODIUM VOID CORES. Barthold, W.P.; Beitel, J.C.; Lam, P.S.K.; Orehwa, Y.; Su, S.F.; Turski, R.B. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 12p. Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

To avoid high energy releases in LMFBR TUC accidents which are accompanied by a failure to scram with a regular shutdown system, various devices have been proposed which would add negative reactivity to the core by either bringing poison material into the core or by creating negative reactivity feedbacks coming from the thermal expansion of the core. While inherent shutdown systems (ISSs) show promise for enhancing safety by adding poison to the reactor, the trigger mechanism and the geometry of the poison are critical design issues. But by postulating an "unprotected" accident, an emergency shutdown capability is denied by definition and the core has to rely solely on inherent safety features. Thermal expansion of core components can lead to the addition of negative reactivity to the core, however, depending on the core restraint system this reactivity margin is usually small and does not prevent sodium boiling. Another approach to enhance the safety of the core is to modify the core design such that the removal of sodium from the core would add only a small amount of positive reactivity or even a negative reactivity. There exists no firm and quantified design goal for the maximum allowable sodium void reactivity. In a simplistic approach, a sodium void reactivity of approximately \$2 or less would be desirable to compensate for roughly \$1 in Doppler reactivity and \$1 as the margin to the prompt-critical stage. But voiding incoherence and fuel motion patterns certainly affect this target value. At this stage of knowledge, it is also not clear if all fuel types would require the same reduction in sodium void reactivity to assure a benign transition phase leading to the termination of a TUC event. A discussion is presented of low sodium void core options which concentrates on the means of lowering sodium void reactivity in general and the oxide fuel form of the uranium-plutonium cycle in particular rather than the carbide, nitride and metal fuel forms.

- 731 (CONF-781022--41) LMFBR SAFETY AND SODIUM BOILING. Hinkle, W.D.; Tschamper, P.M.; Fontana, M.H.; Henry, R.E.; Padilla, A. Jr. (Oak Ridge National Lab., TN (USA); Massachusetts Inst. of Tech., Cambridge (USA); General Electric Co., Sunnyvale, CA (USA); Argonne National Lab., IL (USA); Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract W-7405-ENG-26. 13p. Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

Within the U.S. Fast Breeder Reactor Safety R and D Work Breakdown Structure for Line of Assurance 2, Limit Core Damage, the influence of sodium boiling upon the progression and termination of accidents is being studied in loss of flow, transient overpower, loss of piping integrity, loss of shutdown heat removal system and local fault situations. The pertinent analytical and experimental results of this research to date are surveyed and compared with the requirements for demonstrating the effectiveness of this line of

assurance. A discussion of specific technical issues concerned with sodium boiling and the need for future development work is also presented.

- 732 (CONF-781033--9) HODOSCOPE PERFORMANCE TESTS ON A 91-PIN FUEL BUNDLE AT PARKA. Fink, C.L.; DeVolpi, A.; Rhoades, E.A.; Evans, A.L. (Argonne National Lab., IL (USA); Los Alamos Scientific Lab., NM (USA)). 1978. Contract W-31-109-ENG-38. 6p. Dep. NTIS, PC A02/MF A01.

From IEEE nuclear science symposium; Washington, DC, USA (18 Oct 1978).

Joint LASL/ANL studies of the fuel motion performance capability of a fast-neutron hodoscope have been performed on a 91-pin fuel bundle. Optimization of the neutron detectors used in the hodoscope was also investigated. The neutron detectors tested were a Hornyak button and a stilbene detector with pulse shape discrimination. These studies show that a hodoscope can provide useful fuel motion detection capability for large bundles and that the amount of information provided is greatly increased by the use of neutron detectors with high neutron thresholds and high detection efficiency. These measurements indicate that the TREAT hodoscope can be upgraded to meet most of the fuel motion diagnostic requirements of new safety facilities.

- 733 (CONF-781042--2) STRATEGY FOR THE PRACTICAL UTILIZATION OF THORIUM FUEL CYCLES. Kasten, P.R. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 23p. Dep. NTIS, PC A02/MF A01.

From 5. conference on energy; Rolla, MO, USA (Oct 1978).

There has been increasing interest in the utilization of thorium fuel cycles in nuclear power reactors for the past few years. This is due to a number of factors, the chief being the recent emphasis given to increasing the proliferation resistance of reactor fuel cycles and the thorium cycle characteristic that bred ^{233}U can be denatured with ^{238}U (further, a high radioactivity is associated with recycle ^{233}U , which increases fuel diversion resistance). Another important factor influencing interest in thorium fuel cycles is the increasing cost of U_3O_8 ores leading to more emphasis being placed on obtaining higher fuel conversion ratios in thermal reactor systems, and the fact that thorium fuel cycles have higher fuel conversion ratios in thermal reactors than do uranium fuel cycles. Finally, there is increasing information which indicates that fast breeder reactors have significantly higher capital costs than do thermal reactors, such that there is an economic advantage in the long term to have combinations of fast breeder reactors and high-conversion thermal reactors operating together. Overall, it appears that the practical, early utilization of thorium fuel cycles in power reactors requires commercialization of HTGRs operating first on stowaway fuel cycles, followed by thorium fuel recycle. In the longer term, thorium utilization involves use of thorium blankets in fast breeder reactors, in combination with recycling the bred ^{233}U to HTGRs (preferably), or to other thermal reactors.

- 734 (CONF-781079--4) TRANSIENT HEAT TRANSFER IN CONCRETE. Baker, L. Jr.; Cheung, F.B.; Bingle, J.D. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 12p. Dep. NTIS, PC A02/MF A01.

From Postaccident heat removal information exchange; Ispra, Italy (10 Oct 1978).

The experimental method to study transient heat transfer in concrete has been applied to three concretes of very different compositions. The similarity curves obtained for the three concretes were very much alike. Thermodynamic analysis was used to derive values of the effective volumetric specific heats. The analysis involved the development of idealized compositions to represent the concrete constituents in an approximate way. These were used in conjunction with experimental similarity curves to derive values of effective thermal conductivity for the three concretes which take into account the thermal effects of moisture migration. The results indicated that limestone and magnetite concretes are significantly more conductive at low temperatures than basalt concrete, although all three have values near 1 W/m·K at 700°C.

- 735 (CONF-781105--8) IMPROVED METHOD FOR CONDUCTING FAILED FUEL EXPERIMENTS IN LWR'S. Thomas, R.W.; Klingler, W.B. (EG and G Idaho, Inc., Idaho Falls (USA)). 1978. Contract EY-76-C-07-1570. 5p. Dep. NTIS, PC A02/MF A01. From ANS meeting; Washington, DC, USA (12 Nov 1978).

In conduct of the Sodium Loop Safety Facility (SLSF) experiment program in the Engineering Test Reactor (ETR), destructive tests are performed with up to 37 full length (0.914m fuel zone) LMFBR type fuel pins in a sodium environment. Operation of such experiments in a light water reactor like the ETR requires careful attention to safety. A major problem is how to run destructive experiment transients of sufficient duration to satisfy sponsors needs without jeopardizing the reactor. The paper discusses the way in which EG and G has minimized the difference between nominal and design basis transients, thereby allowing destructive tests of maximum safe duration.

- 736 (CONF-781105--20) FREEZING AND MELTING PHENOMENA IN FINITE-MEDIA, WITH APPLICATION TO MOLTEN FUEL CONTAINMENT FOR SLSF EXPERIMENTS. El-Genk, M.S.; Cronenberg, A.W.; Moore, R.L. (New Mexico Univ., Albuquerque (USA)); EG and G Idaho, Inc., Idaho Falls (USA)). 1978. Contract EY-76-C-07-1570. 12p. Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

The Sodium Loop Safety Facility (SLSF) experiments are being conducted to furnish information concerning the behavior of fast reactor fuel elements under postulated off-normal or hypothesized loss-of-flow accident conditions. Such experiments involve measurements and observations that will permit an assessment of the location and time of sodium void initiation, the nature of sodium expulsion and reentry, the time and nature of cladding failure and motion, the nature of molten fuel motion, and the extent of plugging at the axial extremities of the subassembly. Because such high energy experiments are to be conducted in-pile within the Experimental Test Reactor (ETR) Facility, extensive pretest analysis is required to insure loop integrity. One aspect of this problem is an assessment of molten fuel freezing phenomena on the walls of SLSF to insure the containment capability of the loop, which is the subject of the paper.

- 737 (CONF-781105--57) LOCATION DEPENDENT COMMON CAUSE ANALYSIS WITH AN APPLICATION TO FIRES. Leaver, D.; vonHerrmann, J.; Olmos, J.; Boardman, R. (Science Applications, Inc., Palo Alto, CA (USA)). 1978. Contract EY-77-

C-15-0005. 6p. Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

Portions of document are illegible.

The purpose of the work described is to develop a methodology by which the public risk arising from localized common cause events (e.g., fires) in nuclear plants can be assessed. Attention was paid to maximizing both the completeness and efficiency of the analysis. This methodology was then applied to the analysis of fire events in the Clinch River Breeder Reactor Plant (CRBRP).

- 738 (CONF-781105--58) CONDENSATION FROM A LARGE HCDA VAPOR-GAS BUBBLE ONTO STRUCTURES. Geziak, M.N.; Kress, T.S.; White, J.S. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 6p. Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

A mechanistic assessment of the consequences of a Hypothetical Core Disruptive Accident (HCDA) for a sodium-cooled breeder reactor will require analyses that determine the quantity of fuel released from the primary containment. This release is due to fuel melting, vaporization, and movement through the sodium pool and leakage through the assumed damaged vessel head closure. According to some views, this fuel may move through the cooler sodium pool as a large fuel vapor bubble containing non-condensable fission gases. There is potential for removal of the radionuclide source during this transport by condensation onto structural material (i.e., instrument carriers, control rod drives, support structures, etc.) as well as at the bubble interface. Previous analyses have estimated the condensation rates at the bubble interface. The work described extends a previous model to condensation on structural surfaces in the bubble path. The results of the analysis show that the condensation model for thick steel structures yields similar results as the previous model for condensation on the bubble interface. Due to the high thermal conductivities of both sodium and steel, the mass transfer on the vapor side becomes the controlling factor for condensation. In the case of thin steel structures, the interface temperature rise becomes significant (melting temperatures may be reached) due to the limited heat sink provided by the structure.

- 739 (CONF-781105--66) SLSF DELAYED NEUTRON DETECTION WITH THE ON-LINE SODIUM SAMPLING SYSTEM. Heiberger, W.F.; Holzworth, R.E.; Shire, P.R. (Idaho National Engineering Lab., Idaho Falls (USA)). 1978. Contract EY-76-C-07-1570. 5p. Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

An on-line sodium sampling system (OLSS) is being installed in the Sodium Loop Safety Facility (SLSF) at INEL. The OLSS will provide real time detection of failed fuel pins via fission products deposited in the sodium. The SLSF, now in its fourth test in the Engineering Test Reactor (ETR), permits failed fuel testing under simulated LMFBR transient conditions. The OLSS loop will withdraw a sample stream from the SLSF test loop, and transport it through a delayed neutron monitor (DNM). The oxide control and indication system is provided as a subloop of OLSS to purify the OLSS/SLSF sodium.

- 740 (CONF-781108--), pp 274-295) ANALYSIS OF PROPOSED TRANSIENT OVERPOWER EXPERIMENTS. Campbell, K. (Los Alamos Scientific Lab., NM);

Hurd, D. Jul 1979.

From 4. DOE statistical symposium; Albuquerque, NM, USA (1 Nov 1978).

This problem presentation arose from a project to demonstrate a new approach to the assessment of risk associated with core-disruptive accidents in a liquid metal fast breeder reactor. The project is intended to anticipate problems that may arise when the results of a planned series of experiments are used in this context. A possible model and some simulated test results are presented.

- 741 (CONF-781117--2) U.S. INTEGRAL AND BENCHMARK EXPERIMENTS. Maienschein, F.C. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 21p. Dep. NTIS, PC A02/MF A01.

From FBR shielding seminar; Obninsk, USSR (13 Nov 1978).

Verification of methods for analysis of radiation-transport (shielding) problems in Liquid-Metal Fast Breeder Reactors has required a series of experiments that can be classified as benchmark, parametric, or design-confirmation experiments. These experiments, performed at the Oak Ridge Tower Shielding Facility, have included measurements of neutron transport in bulk shields of sodium, steel, and inconel and in configurations that simulate lower axial shields, pipe chases, and top-head shields. They have also included measurements of the effects of fuel stored within the reactor vessel and of gamma-ray energy deposition (heating). The paper consists of brief comments on these experiments, and also on a recent experiment in which neutron streaming problems in a Gas-Cooled Fast Breeder Reactor were studied. The need for additional experiments for a few areas of LMFBR shielding is also cited.

- 742 (CONF-781117--3) METHODS FOR U.S. SHIELDING CALCULATIONS: APPLICATIONS TO FFTF AND CRBR DESIGNS. Engle, W.W. Jr.; Mynatt, F.R.; Disney, R.K. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 16p. Dep. NTIS, PC A02/MF A01.

From FBR shielding seminar; Obninsk, USSR (13 Nov 1978).

The primary components of the U.S. reactor shielding methodology consist of: (1) computer code systems based on discrete ordinates or Monte Carlo radiation transport calculational methods; (2) a data base of neutron and gamma-ray interaction and gamma-ray-production cross sections used as input in the codes; (3) a capability for processing the cross sections into multigroup or point energy formats as required by the codes; (4) large-scale integral shielding experiments designed to test cross-section data or techniques utilized in the calculations; and (5) a "sensitivity" analysis capability that can identify the most important interactions in a transport calculation and assign uncertainties to the calculated result that are based on uncertainties in all of the input data. The required accuracy for the methodology is to within 5 to 10% for responses at locations near the core to within a factor of 2 for responses at distant locations. Under these criteria, the methodology has proved to be adequate for in-vessel LMFBR calculations of neutron transport through deep sodium and thick iron and stainless steel shields, of neutron streaming through lower axial coolant channels and primary pipe chaseways, and of the effects of fuel stored within the reactor vessel. For ex-vessel LMFBR problems, the methodology requires considerable improvement, the areas of concern including neutron streaming through heating and

ventilation ducts, through the cavity surrounding the reactor vessel, and through gaps around rotating plugs in the reactor heat, as well as gamma-ray streaming through plant shield penetrations.

- 743 (CONF-781117--4) SHIELDING EXPERIMENTAL METHODS. Maienschein, F.C.; Muckenthaler, F.J. (Oak Ridge National Lab., TN (USA)). Nov 1978. Contract W-7405-ENG-26. 13p. Dep. NTIS, PC A02/MF A01.

From FBR shielding seminar; Obninsk, USSR (13 Nov 1978).

Benchmark, parametric, and design-confirmation shielding measurements are made at the Tower Shielding Facility (TSF) in Oak Ridge. A powerful reactor (to 1 MW thermal) is used with spectral modifiers to provide neutron spectra close to those associated with the various parts of an LMFBR. The source strength allows measurements through shields of full reactor thickness using sensitive detectors. The detectors include spectrometers and dosimeters for both neutrons and gamma rays. A large exclusion area provides much flexibility in arranging shield assemblies to be studied.

- 744 (CONF-781117--5) NUCLEAR DATA NEEDS FOR FAST BREEDER REACTOR SHIELDING. Oblow, E.M.; Perey, F.G. (Oak Ridge National Lab., TN (USA)). Nov 1978. Contract W-7405-ENG-26. 16p. Dep. NTIS, PC A02/MF A01.

From FBR shielding seminar; Obninsk, USSR (13 Nov 1978).

A review of neutron and gamma-ray cross section data needs for fast reactor shielding is presented in light of the recent advances made in assessing these needs through sensitivity studies. Total and partial cross sections and energy and angular distribution data for neutrons are surveyed as well as gamma-ray production cross sections. The strengths and deficiencies of currently available benchmark-quality integral experiments are also discussed with respect to their use in creating adjusted cross section libraries for design work. The availability of first round covariance data in ENDF/B-IV and plans for ENDF/B-V are also reviewed. This latter information makes it possible to quantitatively assess the quality of current cross section data libraries and also puts adjustment and data assessment procedures on a firmer basis.

- 745 (CONF-781117--6) INVESTIGATION OF STEEL-SODIUM-IRON SHIELDS. Oblow, E.M.; Maerker, R.E. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 15p. Dep. NTIS, PC A02/MF A01.

From FBR shielding seminar; Obninsk, USSR (13 Nov 1978).

An analysis of experimental data from 21 fast reactor shield configurations containing steel, sodium, and iron were made as part of a study of the upper axial shielding needs of the Clinch River Breeder Reactor. The measured data were analyzed using both one- and two-dimensional discrete ordinates transport codes and several cross section libraries based on ENDF/B-IV data with group structures of 51 and 171 neutron groups. One-dimensional sensitivity studies using the 171 group library and ENDF/B-IV covariance files for sodium and iron data were used to determine the sensitivities of the measured data to multigroup cross sections and to estimate uncertainties in the calculated results. Results indicate that the standard 51-group design cross section library could be expected to predict the measurements to within 30% over 12 decades of attenuation although a few of the deepest penetration configurations

showed disagreements as large as a factor of three. The sensitivity results revealed very high sensitivity of the measurements to total cross section minima and cross sections from 5 to 10 MeV in sodium and iron in the deep penetration configurations. As a result, large uncertainties in the calculated results arose from small uncertainties in the cross section data. These results indicate the need for better measurements of the total cross section minima in sodium, especially around 300 keV.

- 746 (CONF-781117--7) DESIGN EXPERIENCE: CRBRP RADIATION SHIELDING. Disney, R.K.; Chan, T.C.; Gallo, F.G.; Hedgecock, L.R.; McGinnis, C.A.; Wrights, G.N. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Nov 1978. Contract EY-76-C-15-2395. 27p. Dep. NTIS, PC A03/MF A01.

From FBR shielding seminar; Obninsk, USSR (13 Nov 1978).

The Clinch River Breeder Reactor Plant (CRBRP) is being designed as a fast breeder demonstration project in the U.S. Liquid Metal Fast Breeder Reactor (LMFBR) program. Radiation shielding design of the facility consists of a comprehensive design approach to assure compliance with design and government regulatory requirements. Studies conducted during the CRBRP design process involved the aspects of radiation shielding dealing with protection of components, systems, and personnel from radiation exposure. Achievement of feasible designs, while considering the mechanical, structural, nuclear, and thermal performance of the component or system, has required judicious trade-offs in radiation shielding performance. Specific design problems which have been addressed are in-vessel radial shielding to protect permanent core support structures, flux monitor system shielding to isolate flux monitoring systems for extraneous background sources, reactor vessel support shielding to allow personnel access to the closure head during full power operation, and primary heat transport system pipe chaseway shielding to limit intermediate heat transport system sodium system coolant activation. The shielding design solutions to these problems defined a need for prototypic or benchmark experiments to provide assurance of the predicted shielding performance of selected design solutions and the verification of design methodology. Design activities of CRBRP plant components and systems, which have the potential for radiation exposure of plant personnel during operation or maintenance, are controlled by a design review process related to radiation shielding. The program implements design objectives, design requirements, and cost/benefit guidelines to assure that radiation exposures will be "as low as reasonably achievable" (ALARA).

- 747 (CONF-781181--2) PREDICTION OF THERMAL CONSEQUENCES OF PLENUM FISSION GAS RELEASE IN AN LMFBR FUEL ASSEMBLY. Kadambi, N.P. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1978. Contract EY-76-C-15-2395. 5p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1978).

One of the types of hypothetical accidents important in the safety analysis of Liquid Metal Fast Breeder Reactors is that in which fission gas release is postulated to occur from the plena of fuel pins at such a rate that significant perturbation to coolant flow occurs. Past consideration of such events has employed highly conservative calculations in the interest of simplicity and expediency. This

paper describes a method developed to predict more accurately the thermal consequences of such fission gas release events.

- 748 (CONF-781202--25) EFFECTS OF NOTCHES ON ELEVATED-TEMPERATURE, LOW-CYCLE FATIGUE BEHAVIOR OF TYPE 304 STAINLESS STEEL. Raske, D.T.; Maiya, P.S. (Argonne National Lab., IL (USA)). Dec 1978. Contract W-31-109-ENG-38. 23p. Dep. NTIS, PC A02/MF A01.

From ASME meeting; San Francisco, CA, USA (10 Dec 1978).

The results of an investigation into the effects of geometric stress concentrations on the elevated-temperature low-cycle fatigue behavior of Type 304 stainless steel are presented. The objective of this study was to develop a data base that could be used to verify the ASME Code Case 1592-8 design method for predicting the creep-fatigue behavior of structural components which contain discontinuities. In continuous-cycling tests, the local strains at the notch root determined from the Code Case procedure result in fatigue-life estimates that are conservative when compared to the experimental values. For tests containing a hold time at the peak tensile strain, the Code Case procedure results in local strains that are nearly identical to the experimentally obtained local strains, if the local stress is assumed to relax to the nominal value during the hold time.

- 749 (CONF-781202--26) ACOUSTICS AND VOIDING DYNAMICS DURING SLSF SIMULATIONS OF LMFBR UNDERCOOLING TRANSIENTS. Anderson, T.T.; Kuzay, T.M.; Marr, W.W.; Miles, K.J.; Pedersen, D.R.; Thompson, D.H.; Wilson, R.E. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 26p. Dep. NTIS, PC A03/MF A01.

From ASME meeting; San Francisco, CA, USA (10 Dec 1978).

The SLSF is the largest U.S. in-reactor test vehicle for steady-state and transient experiments in an environment typical of a LMFBR core. The SLSF experiment program, sponsored by the Department of Energy, contributes to the LMFBR safety assurance program by providing data on key phenomena that occur during postulated reactor accidents. This paper describes completed SLSF experiments, in-core instrumentation used, and methods of data interpretation to determine sodium boiling and voiding dynamics. Boiling inception is shown to be identifiable from several types of in-core instruments. Location of the boiling front and void growth derived from experimental data are compared with analytical predictions. These and other data form the basis to improve understanding of accidents and to validate or guide the development of accident analysis methods.

- 750 (CONF-781219--1) TENSILE, CREEP, AND CREEP--RUPTURE STUDIES ON 2 1/4 CR--1 MO STEEL. Klueh, R.L. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 17p. Dep. NTIS, PC A02/MF A01.

From US/UKAEA specialists meeting on mechanical properties; Oak Ridge, TN, USA (4 Dec 1978).

Tensile, creep, and creep-rupture studies were made on annealed or isothermally annealed air-melted 2 1/4 Cr--1 Mo steel. Creep strength is the result of the interaction of solid solution hardening and dispersion strengthening. Microstructure is primarily proeutectoid ferrite. (FS)

- 751 (CONF-781219--4) JOINING TECHNOLOGY DEVELOPMENT SUMMARY, 189A OH024. Edmonds, D.P. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 34p. Dep. NTIS, PC A03/MF A01.
From US/UKAEA specialists meeting on mechanical properties; Oak Ridge, TN, USA (4 Dec 1978).
Joining technology is discussed for Breeder Reactor technology including welding development and mechanical properties of weldments for austenitic stainless steels, high-temperature structural design data for austenitic stainless steel weldments and advanced transition joints is discussed. Technology developed in this program is briefly reviewed. (FS)
- 752 (CONF-781219--5) OVERVIEW OF U.S. EFFORTS DIRECTED TOWARDS DEVELOPMENT OF MECHANICAL PROPERTIES DATA IN SUPPORT OF FBR DEVELOPMENT. Brinkman, C.R. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 14p. Dep. NTIS, PC A02/MF A01.
From US/UKAEA specialists meeting on mechanical properties; Oak Ridge, TN, USA (4 Dec 1978).
The object of U.S. Breeder Reactor Program associated with primary and secondary circuit structural materials is to develop a design data base and associated design technology on commercially available material as well as new alloys. The commercially available materials include ASIA 316 and 304. Chromium--molybdenum steels and alloy 800H. (FS)
- 753 (CONF-790202-- , pp 9-11) CLINCH RIVER BREEDER REACTOR. Baron, S. Jul 1979.
From 6. annual energy conference and exhibition energy and the public; Knoxville, TN, USA (21 Feb 1979).
Mr. Baron says the administration's effort to terminate the Clinch River Breeder Reactor (CRBR) project is symptomatic; they have also placed restrictions on fusion, coal, solar, and other areas of energy development in which technological advances are held back in order to force conservation. Because the breeder reactor, unlike solar and fusion energy, is both economically and technically feasible, a demonstration plant is needed. The contentions that the CRBR design is obsolete, that its proposed size is inappropriate, or that plutonium can be diverted for weapons proliferation are argued to be invalid. Failure to complete the CRBR will have both economic and national security repercussions.
- 754 (CONF-790236-- , pp 32-40) LIQUID-METAL FAST BREEDER REACTOR STRUCTURAL MATERIALS DESIGN CONSIDERATIONS. Huddleston, R.L. (Oak Ridge National Lab., TN). Jun 1979.
From Workshop on time-dependent fracture of materials; Germantown, MD, USA (15 Feb 1979).
This paper gives a brief overview of the LMFBR, to describe its key components, addresses two key structural problems, reviews high-temperature materials utilized, and places bounds on expected operating conditions. The current status of materials utilization in the LMFBR is summarized as follows: with the exception of the reactor upper internals, design needs for the LMFBR can be met with currently approved Code materials; Inconel 718 can potentially solve the thermal striping problems in the reactor upper internals; temperature, stress-strain levels, and design lifetime of the LMFBR push currently approved Code materials toward their limits of usefulness.
- 755 (CONF-790236--1) LIQUID METAL FAST BREEDER REACTOR STRUCTURAL MATERIALS DESIGN CONSIDERATIONS. Huddleston, R.L. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 18p. Dep. NTIS, PC A02/MF A01.
From Workshop on time-dependent fracture of materials; Germantown, MD, USA (15 Feb 1979).
In summary, this paper has attempted to give a brief overview of the LMFBR, to describe its key components, to address two key structural problems, to review high-temperature materials utilized, and finally, to attempt to place bounds on expected operating conditions. The current status of materials utilization in the LMFBR might be summarized as follows: (1) With the exception of the reactor upper internals, design needs for the LMFBR can be met with currently approved Code materials. (2) Inconel 718 can potentially solve the thermal striping problems in the reactor upper internals. (3) Temperature, stress-strain levels, and design lifetime of the LMFBR push currently approved Code materials toward their limits of usefulness.
- 756 (CONF-790239--1) COMPARISON OF NUMERICAL RESULTS WITH EXPERIMENTAL DATA FOR SINGLE-PHASE NATURAL CONVECTION IN AN EXPERIMENTAL SODIUM LOOP. Ribando, R.J. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 45p. Dep. NTIS, PC A03/MF A01.
From Specialists' meeting on decay heat removal and natural convection in FBRs; Upton, Long Island, NY, USA (28 Feb 1979).
A comparison is made between computed results and experimental data for a single-phase natural convection test in an experimental sodium loop. The test was conducted in the Thermal-Hydraulic Out-of-Reactor Safety (THORS) facility, an engineering-scale high temperature sodium loop at the Oak Ridge National Laboratory (ORNL) used for thermal-hydraulic testing of simulated Liquid Metal Fast Breeder Reactor (LMFBR) subassemblies at normal and off-normal operating conditions. Heat generation in the 19 pin assembly during the test was typical of decay heat levels. The test chosen for analysis in this paper was one of seven natural convection runs conducted in the facility using a variety of initial conditions and testing parameters. Specifically, in this test the bypass line was open to simulate a parallel heated assembly and the test was begun with a pump coastdown from a small initial forced flow. The computer program used to analyze the test, LONAC (LOW flow and NATural Convection) is an ORNL-developed, fast-running, one-dimensional, single-phase, finite-difference model used for simulating forced and free convection transients in the THORS loop.
- 757 (CONF-790306--5) PERFORMANCE OF EBR-II MARK-II METALLIC DRIVER FUEL UP TO 6750°C. Einziger, R.E. (Argonne National Lab., IL (USA)). Mar 1979. Contract W-31-109-ENG-38. 17p. Dep. NTIS, PC A02/MF A01.
From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).
The performance of EBR-2 Mark-II fuel at 6750°C is significantly different from that at 5900°C. Despite the differences, the elements were all successfully irradiated to 8 at. % burnup before being removed from the reactor unbreached. Some of the more significant differences in performance at the higher temperatures are (1) The fuel-element cladding showed a double-peaking strain profile, with enhanced swelling and creep due to extensive

carbide precipitation, an FCCI enhanced fission gas stress, or FCMI. (2) The FCCI zone in the fuel was cracked and represented 25% of the cladding thickness. This zone was substantially larger than observed at 590°C. (3) The fuel pin lifted 12 mm to the restrainer dimple, so some type of restrainer may be necessary in future designs. A restrainer had previously been deemed unnecessary based on irradiations at 590°C. (4) The fission-gas release is less at a fuel-centerline temperature of 720°C compared at 650°C. This difference is primarily due to the upper third of the element having very little open porosity. Owing to the lack of bond sodium infiltration into the fuel at the higher temperature, the fission gas pressure is the same at both temperatures.

- 758 (CONF-790306--16) PERFORMANCE OF U--PU--ZR METAL FUEL IN 1000 MWE LMFBRs. Lam, P.S.K.; Turski, R.B.; Barthold, W.P. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 15p. Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

The present analyses indicate that very conservatively designed U--Pu--Zr metal-fueled 1000 MW(e) LMFBRs have CSDT in the range of 9 to 10 years, with corresponding specific inventories ranging from 2.5 to 3.5 kg/MWe. Doubling times in the range of 8 years are achievable with only minor changes in the fuel pin design or operating conditions.

- 759 (CONF-790361--1) FUEL AND COOLANT MOTIONS FOLLOWING PIN FAILURE: EPIC MODELS AND THE PBE-5S EXPERIMENT. Garner, P.L.; Abramson, P.B. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 14p. Dep. NTIS, PC A02/MF A01.

From LMFBR workshop; Albuquerque, NM, USA (12 Mar 1979).

The EPIC computer code has been used to analyze the post-fuel-pin-failure behavior in the PBE-5S experiment performed at Sandia Laboratories. The effects of modeling uncertainties on the calculation are examined. The calculations indicate that the majority of the piston motion observed in the test is due to the initial pressurization of the coolant channel by fuel vapor at cladding failure. A more definitive analysis requires improvements in calculational capabilities and experiment diagnostics.

- 760 (CONF-790363--1) OVERVIEW OF US EFFORTS TO VALIDATE ANALYSIS METHODS AND DESIGN CRITERIA. Corum, J.M. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 39p. Dep. NTIS, PC A03/MF A01.

From US Japan specialist meeting on high temperature piping design and analysis; Pittsburgh, PA, USA (26 Mar 1979).

As a part of its basic breeder reactor technology activities, the United States has a High-Temperature Structural Design Program aimed at establishing, and transferring to designers a rationally sound and experimentally validated structural design technology that will assure freedom from structural failures. Both development and validation efforts are centered on three main ingredients of the technology: (1) mathematical descriptions (constitutive equations) of the deformation behavior of the alloys of interest, (2) time-dependent rupture, or cracking, models, again for the alloys of interest, and (3) detailed design analysis methods and failure criteria based on these inputs. The objective of this

presentation is to provide a brief overview of the status of efforts to experimentally assess and validate each of these three technology ingredients. In the case of deformation and rupture models, the current procedures and rules are outlined, and their significant features are assessed through comparisons with results from uniaxial and multiaxial material behavior tests. Design analysis methods and failure criteria are, on the other hand, discussed relative to the results of experimental benchmark structural tests. Although several significant needs and problem areas remain, data generated to date tend to provide limited validation of key aspects of the existing technology. Current constitutive equations represent many of the observed behavioral features; limited available data support current time-dependent rupture criteria reasonably well; and it is observed that structural analysis predictions capture most of the experimentally observed features of inelastic structural behavior.

- 761 (CONF-790363--2) EVALUATION OF THE INFLUENCE OF SEISMIC RESTRAINT CHARACTERISTICS ON LMFBR PIPING. Mallett, R.H. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract EY-76-C-15-2395. 21p. Dep. NTIS, PC A02/MF A01.

From US Japan specialist meeting on high temperature piping design and analysis; Pittsburgh, PA, USA (26 Mar 1979).

For the Clinch River Breeder Reactor Plant (CRBRP) heat transport system piping within the reactor containment building, dynamic analyses of the piping loops have been performed to study the effect of restraint stiffness on the dynamic behavior of the piping, i.e., changes in frequencies, mode shapes, loads, stresses, and displacements. In addition, analysis and testing of typical CRBRP restraint system components have been performed for the purpose of quantifying and verifying the basic characteristics of the restraints used in the piping system dynamic analysis. The results are presented of the analysis and testing described above. In particular, it is shown that piping restraint effects on breeder reactor piping response are important considerations in the design/analysis process.

- 762 (CONF-790363--3) LOAD-TIME HISTOGRAM AND ELASTIC ANALYSIS OF CRBRP HEAT TRANSPORT SYSTEM PIPING. Mello, R.M. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract EY-76-C-15-2395. 21p. Dep. NTIS, PC A02/MF A01.

From US Japan specialist meeting on high temperature piping design and analysis; Pittsburgh, PA, USA (26 Mar 1979).

The CRBRP heat transport system piping was designed and analyzed for the operating and load conditions as specified in the Piping Design Specification. The various types of design loading conditions have been categorized into design, normal, upset, emergency and faulted conditions, and organized into a load histogram. Using the thermal transient defined for each condition and the histogram, a set of load cycles was constructed using the 863 startup-shutdown conditions. The stress analysis of the piping was performed on the basis of the thermal transients and loadings resulting from the loading categories and the histogram. The ELTEMP computer program was used to perform the elastic ASME Code compliance analysis. The input required to perform the Code evaluation included the pipe cross-section geometry, component type (elbow, tee, straight, etc.), the forces and moments output data from

the flexibility analyses, average temperature, linear temperature gradients, non-linear temperature gradients, pressure, etc.

- 763 (CONF-790363--4) APPLICATION OF ASME CODE CASE N-47 CONCEPTS TO CRBRP PRIMARY PIPING. Mallett, R.H. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract EY-76-C-15-2395. 22p. Dep. NTIS, PC A02/MF A01.

From US Japan specialist meeting on high temperature piping design and analysis; Pittsburgh, PA, USA (26 Mar 1979).

Portions of document are illegible.

Section III of the ASME Code provides rules in Subsection NB-3600 which greatly simplify the structural evaluation of piping. The key to this simplification is in the use of stress analysis formulas based upon stress indices. These formulas are applicable in the evaluation of compliance with limits on elastically calculated quantities. At present, Code Case N-47 permits, but does not specify, the use of analyses based upon piping stress indices. This paper describes a set of formulas for piping analysis using stress indices. Specifically, stress intensities needed for evaluation of compliance with Code Case N-47 limits on elastically calculated are given using piping stress indices. The specified formulas are examined by comparison with related ANSI B31.1 rules and test data.

- 764 (CONF-790363--5) DESIGN CONSIDERATIONS FOR CRBRP PRIMARY PIPING SYSTEM OPERATING AT ELEVATED TEMPERATURES. Mello, R.M. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract EY-76-C-15-2395. 18p. AT.

From US Japan specialist meeting on high temperature piping design and analysis; Pittsburgh, PA, USA (26 Mar 1979).

The heat transport system sodium piping for the Clinch River Breeder Reactor Plant (CRBRP) within the reactor containment building must withstand high temperatures for long periods of time. Each phase of the mechanical design process of the piping system is influenced by elevated temperature considerations which include material thermal creep effects, ratchetting caused by rapid temperature transients and stress relaxation, and material degradation effects. This requires that careful attention be given to through-wall temperature gradients and time-dependent material behavior in the design and analysis process. The description of the structural design philosophy and approaches taken to design the CRBRP piping operating in a high temperature environment are presented. In addition, the resulting design of the heat transport system piping is discussed along with the special features that resulted from the elevated temperature considerations.

- 765 (CONF-790370--1) STUDY OF STRUCTURAL ATTACHMENTS OF A POOL TYPE LMFBR VESSEL THROUGH SEISMIC ANALYSIS OF A SIMPLIFIED THREE DIMENSIONAL FINITE ELEMENT MODEL. Ahmed, H.; Ma, D. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 15p. Dep. NTIS, PC A02/MF A01.

From 5. international conference on structural mechanics in reactor technology; Berlin, F.R. Germany (20 Mar 1979).

A simplified three dimensional finite element model of a pool type LMFBR in conjunction with the computer program ANSYS is developed and scoping results of seismic analysis are produced. Through this study various structural attachments of a pool type LMFBR like the reactor vessel skirt support,

the pump support and reactor shell-support structure interfaces are studied. This study also provides some useful results on equivalent viscous damping approach and some improvements to the treatment of equivalent viscous damping are recommended. This study also sets forth pertinent guidelines for detailed three dimensional finite element seismic analysis of pool type LMFBR (to be published in a separate paper).

- 766 (CONF-790374--1) RADIOACTIVE WASTE MANAGEMENT AT A LIQUID METAL FAST BREEDER REACTOR. Abrams, C.S.; Fryer, R.H.; Witbeck, L.C. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 17p. Dep. NTIS, PC A02/MF A01.

From AECD-IAEA symposium on the on-site management of power reactor; Zurich, Switzerland (26 Mar 1979).

This paper presents the radioactive waste production and management at a Liquid Metal Fast Breeder Reactor-II (EBR-II), which is operated for the US Department of Energy by the Argonne National Laboratory at the Idaho National Engineering Laboratory (INEL). Since this facility, in addition to supplying power has been used to demonstrate the breeder, fuel cycling, and recently operations with defective fuel elements, various categories of waste have been handled safely over some 14 years of operation. Liquid wastes are processed such that the resulting effluent can be discharged to an uncontrolled area. Solid wastes up to 10,000 R/hr are packaged and shipped contamination-free to a disposal site or interim storage with exposures to personnel approximately 10 mrem. Gaseous waste discharges are low such as 143 Ci of noble gases in 1978 and do not have a significant effect on the environment even with operations with breached fuel.

- 767 (CONF-790402--16) ELASTIC-PLASTIC-CREEP ANALYSIS OF SHELLS. Pai, D.H. (Foster Wheeler Corp., Livingston, NJ (USA)). 1979. Contract EW-76-C-2395. 23p. Dep. NTIS, PC A02/MF A01.

From Computational methods in nuclear engineering; Williamsburg, VA, USA (23 Apr 1979).

This paper presents the recent experience of a designer/fabricator of nuclear heat transport components in the area of elastic-plastic-creep analysis of shell-like structures. A brief historical perspective is first given to highlight the evolution leading to the present industry practice. The ASME elevated temperature design criteria will be discussed followed by examples of actual computations performed to support the design/analysis and fabrication of a breeder reactor component in which a substantial amount of elastic-plastic-creep analysis was performed. Mathematical challenges encountered by the design analyst in these problems will be highlighted. Developmental needs and future trends will then be given.

- 768 (CONF-790402--18) VENUS-III TWO-DIMENSIONAL MULTI-COMPONENT THERMAL HYDRAULIC TECHNIQUES. Weber, D.P. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 19p. Dep. NTIS, PC A02/MF A01.

From Computational methods in nuclear engineering; Williamsburg, VA, USA (23 Apr 1979).

In recent analyses of the initiating phase in LMFBR core disruptive accidents the energy deposition rate may not be nearly so high as originally thought and the development of

material motion and interaction may take place on a time scale considerably larger than the classic disassembly time scale of milliseconds. This introduces a considerably different twist to the problem and it becomes apparent that processes heretofore ignored, such as differential motion and heat exchange, may become important. In addition, time scales may become long enough that substantial core material motion may take place and since rearrangement in more critical configurations cannot be absolutely precluded, capability for extended motion analysis, not easily performed with Lagrangian techniques in multi-dimensions, become desirable. Such considerations provided the motivation for developing a hydrodynamic algorithm to resolve these questions, and an Eulerian rather than Lagrangian frame of reference was chosen, primarily to handle extended motion and interpenetration. The results of the study are described.

- 769 (CONF-790402--19) NODAL METHOD FOR FAST REACTOR ANALYSIS. Shober, R.A. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 16p. Dep. NTIS, PC A02/MF A01.

From Computational methods in nuclear engineering; Williamsburg, VA, USA (23 Apr 1979).

In this paper, a nodal method applicable to fast reactor diffusion theory analysis has been developed. This method has been shown to be accurate and efficient in comparison to highly optimized finite difference techniques. The use of an analytic solution to the diffusion equation as a means of determining accurate coupling relationships between nodes has been shown to be highly accurate and efficient in specific two-group applications, as well as in the current multigroup method.

- 770 (CONF-790402--20) APPLICATION OF AN LP MODEL TO BREEDER STRATEGY STUDIES. Till, C.E.; Chang, Y.I. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 17p. Dep. NTIS, PC A02/MF A01.

From Computational methods in nuclear engineering; Williamsburg, VA, USA (23 Apr 1979).

The paper discusses the relationships between the capital cost differential (FBR--LWR) allowable for economic breeder introduction and energy demand, resource availability (through price--quantity schedule), and economic environment for a range of future projections. The ALPS linear programming reactor systems analysis code, developed by Hanford Engineering Development Laboratory, was used for economic optimizations where they were done, and where they were not it provided a useful tool to compute the discounted total system power cost over the planning horizon for a given set of reactor mix and cost parameters.

- 771 (CONF-790404--), pp 145-170) FFTF IMPLEMENTATION OF TRAINING REQUIREMENTS FOR GOVERNMENT AND COMMERCIALY OWNED FACILITIES. Gaden, A.D. (Hanford Engineering Development Lab., Richland, WA); Izatt, R.D.; Wiborg, J.C. Apr 1979.

From 3. symposium on the training on nuclear facility personnel; Gatlinburg, TN, USA (29 Apr 1979).

The FFTF Training Program is a specific example of how training guidelines provided by NRC can be implemented in principle in a DOE owned reactor. The DOE requirements for government owned facilities and the NRC requirements for commercially owned facilities

are compared to components of the FFTF Training program. These components will be described, actual status provided and evaluated as to meeting requirements. Cold Plant Qualification (equivalent to NRC Cold Plant Licensing) is the next major milestone for FFTF Operator training. Additional requirements or constraints such as operator time utilization, recruiting and turnover, plant availability and startup testing greatly affect the ability to meet this milestone. These constraints and requirements and the resulting compromises will be evaluated relative to meeting DOE requirements.

- 772 (CONF-790408--3) STRENGTH PROPERTIES OF CONCRETE AT ELEVATED TEMPERATURES. Freskakis, G.N.; Burrow, R.C.; Debbas, E.B. (Burns and Roe, Inc., Oradell, NJ (USA)). 1979. Contract EY-76-C-03-1145. 30p. Dep. NTIS, PC A03/MF A01.

From 3. ASCE specialty conference on the structural design of nuclear plant facilities; Boston, MA, USA (2 Apr 1979).

A study is presented concerning the compressive strength, modulus of elasticity, and stress-strain relationships of concrete at elevated temperatures. A review of published results provides information for the development of upper and lower bound relationships for compressive strength and the modulus of elasticity and establishes exposure conditions for a lower bound thermal response. The relationships developed from the literature review are confirmed by the results of a verification test program. The strength and elasticity relationships provide a basis for the development of design stress-strain curves for concrete exposed to elevated temperatures.

- 773 (CONF-790602--36) WHOLE-CORE THERMAL-HYDRAULIC TRANSIENT CODE DEVELOPMENT AND VERIFICATION FOR LMFBR ANALYSIS. Spencer, D.R. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Apr 1979. Contract EW-76-C-15-2395. 5p. Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

Predicted performance during both steady state and transient reactor operation determines the steady state operating limits on LMFBRs. Unnecessary conservatism in performance predictions will not contribute to safety, but will restrict the reactor to more conservative, less economical steady state operation. The most general method for reducing analytical conservatism in LMFBR's without compromising safety is to develop, validate and apply more sophisticated computer models to the limiting performance analyses. The purpose of the on-going Natural Circulation Verification Program (NCVP) is to develop and validate computer codes to analyze natural circulation transients in LMFBRs, and thus, replace unnecessary analytical conservatism with demonstrated calculational capability.

- 774 (CONF-790602--52) THEORETICAL EVALUATION OF EX-VESSEL MONITORING FOR INITIAL FUEL LOADING OF A LIQUID METAL FAST BREEDER REACTOR. Mihalcz, J.T.; Selby, D.L.; Knee, H.E. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 7p. Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

Transport theory calculations were used to determine the feasibility of monitoring the fuel loading to initial criticality of the Clinch River Breeder Reactor (CRBR) with a

detector in a cavity outside the reactor vessel. Such monitoring of the CRBR with an ex-vessel detector will be different from monitoring of previous LMFBRs, where in-vessel detectors were used. The feasibility of ex-vessel monitoring will depend mainly on two criteria: (1) sensitivity - will there be enough counts to obtain adequate counting statistics; and (2) interpretability - will the count rate obtained during the initial fuel loading sequence be sufficient to determine the neutron multiplication or reactivity. Satisfying these criteria will assure that the reactor can be loaded safely to initial criticality. The sensitivity criterion can be satisfied by inserting an additional neutron source (one much more intense than the inherent neutron source of the fuel subassemblies) into the core center and using ex-vessel detectors with high sensitivity, such as multiple BF₃ counters mounted in a graphite moderator block. These calculations were used to determine the intensity of the additional source required to produce adequate counting rates at the ex-vessel detectors.

- 775 (CONF-790602--59) DETERMINATION OF CURRENT WEIGHTED TOTAL CROSS SECTIONS IN THE UNRESOLVED ENERGY REGION. Barhen, J.; Cacuci, D.G. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 7p. Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

As part of the preliminary ENDF/B-5 data testing, recommendations were made to reassess the computation of the effective point-wise current-weighted cross sections in the unresolved energy region. No previously published work has taken fully into account such important effects as interference scattering and within-sequence overlap. In addition, the extension of the unresolved region to higher energies in ENDF/B-5, in order to include competitive widths, increases the computational complexity. Because of the potential cost penalties associated with design uncertainties in large LMFBR's, it was necessary to develop a procedure which takes these effects fully into account. The new methodology presented here is to be used in the generation of multigroup libraries based on the final version of ENDF/B-5. 1 table. (RWR)

- 776 (CONF-790602--60) INTERLABORATORY COMPUTATIONAL COMPARISONS OF CRITICAL FAST TEST REACTOR PIN LATTICES. Mincey, J.F.; Kerr, H.T.; Durst, B.M. (Oak Ridge National Lab., TN (USA)); Battelle Pacific Northwest Labs., Richland, WA (USA)). 1979. Contract W-7405-ENG-26. 8p. Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

An objective of the Consolidated Fuel Reprocessing Program's (CFRP) nuclear engineering group at Oak Ridge National Laboratory (ORNL) is to ensure that chemical equipment components designed for the reprocessing of spent LMFBR fuel (among other fuel types) are safe from a criticality standpoint. As existing data are inadequate for the general validation of computational models describing mixed plutonium-uranium oxide systems with isotopic compositions typical of LMFBR fuel, a program of critical experiments has been initiated at the Battelle Pacific Northwest Laboratories (PNL). The first series of benchmark experiments consisted of five square-pitched lattices of unirradiated Fast Test Reactor (FTR) fuel moderated and reflected by light water. Calculations of these five experiments have been conducted by both ORNL/

CFRP and PNL personnel with the purpose of exploring how accurately various computational models will predict $k_{\text{sub eff}}$ values for such neutronic systems and if differences between $k_{\text{sub eff}}$ values obtained with these different models are significant.

- 777 (CONF-790602--61) INFLUENCE OF FISSION SPECTRA UNCERTAINTIES ON CALCULATED FAST REACTOR PARAMETERS. Lucius, J.L.; Marable, J.H. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 6p. Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

The effect of fission spectra uncertainties on calculated fast reactor parameters is studied by (1) considering the sensitivities to fission spectrum parameters of four performance parameters in the fast reactor benchmark ZPR-6/7, (2) estimating the uncertainties in these calculated performance parameters due to uncertainties in the fission spectra, (3) reporting the change in ZPR-6/7 calculated performance parameters due to fission spectra changes going from ENDF/B-IV to proposed ENDF/B-VP, and (4) determining what fast benchmark integral experiments are trying to tell us about ENDF/B-IV fission spectrum parameters as variables in a least squares adjustment procedure.

- 778 (CONF-790602--62) SODIUM COMPONENT RELIABILITY DATA COLLECTION AT CREDO. Bott, T.F.; Haas, P.M.; Manning, J.J. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 6p. Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

The Centralized Reliability Data Organization (CREDO) has been established at Oak Ridge National Laboratory (ORNL) by the Department of Energy to provide a national center for collection, evaluation and dissemination of reliability data for advanced reactors. While the system is being developed and continuous data collection at the two U.S. reactor sites (EBR-II and FFTF) is being established, data on advanced reactor components which have been in use at U.S. test loops and experimental reactors have been collected and analyzed. Engineering, operating and event data on sodium valves, pumps, flow meters, rupture discs, heat exchangers and cold traps have been collected from more than a dozen sites. The results of analyses of the data performed to date are presented.

- 779 (CONF-790602--63) COMPARISON OF THE HAARM-3 FALLOUT MODEL WITH NUCLEAR SAFETY PILOT PLANT (NSPP) DATA. Han, J.T.; Kress, T.S.; Adams, R.E. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 6p. Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

Sodium-oxide aerosols have been generated in the Nuclear Safety Pilot Plant (NSPP) experiment under vessel conditions appropriate for the LMFBR secondary containment following an HCDA. The primary goal of the NSPP program is to provide experimental validation for the HAARM-3 aerosol behavioral code. Validation can be determined by direct comparison of code predictions and integral test results using the same initial and boundary conditions. In addition, separate physical models within the code (e.g., for agglomeration, plate-out, and gravitational fallout) may be validated by a comparison of predicted values with measured instantaneous values. A comparison is presented

for the gravitational fallout rate which is considered to be the primary depletion mechanism under prototypic secondary containment conditions.

- 780 (CONF-790602--65) EVALUATION OF STOCHASTIC VOID MODELS FOR BUBBLE WORTH CALCULATIONS. Vaughan, E.U.; Hoffman, T.J. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group; Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 7p. Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

The reactivity due to the collapse of a large number of small bubbles in a reactor core has been analyzed with Behrens' method, Benoist's method, and the probability table method. Each analysis has used a different stochastic model for the bubble distribution within the reactor core: the Nicholson--Goldsmith calculations have employed a closely-spaced holes model; the Gelbard--Lell calculations have employed a random cube model; the Hoffman--Petric calculations have employed a nuclear model. However, any of the models can be used with any of the methods. In this paper, the three void models are evaluated. Based on this analysis, the closely-spaced holes model is recommended as the most suitable for analysis of bubble collapse in a molten, LMFBR core.

- 781 (CONF-790602--87) ANALYTICAL METHOD TO ACCURATELY PREDICT LMFBR CORE FLOW DISTRIBUTION. Carelli, M.D.; Willis, J.M. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract EY-76-C-15-2395. 10p. Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

An accurate and detailed representation of the flow distribution in LMFBR cores is very important as the starting point and basis of the thermal and structural core design. Previous experience indicated that the steady state and transient core design is as good as the core orificing; thus, a new orificing philosophy satisfying a priori all design constraints was developed. However, optimized orificing is a necessary, but not sufficient condition for achieving the optimum core flow distribution, which is affected by the hydraulic characteristics of the remainder of the primary system. Consequently, an analytical model of the overall primary system was developed, resulting in the CATFISH computer code, which, even though specifically written for LMFBRs, can be used for any reactor employing ducted assemblies.

- 782 (CONF-790615--11) STRUCTURAL ANALYSIS OF THE UPPER INTERNALS STRUCTURE FOR THE CLINCH RIVER BREEDER REACTOR PLANT. Houtman, J.L. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract EW-76-C-15-2395. 54p. Dep. NTIS, PC A04/MF A01.

From 3. US national congress on pressure vessels and piping; San Francisco, CA, USA (25 Jun 1979).

The Upper Internals Structure (UIS) of the Clinch River Breeder Reactor Plant (CRBRP) provides control of core outlet flow to prevent severe thermal transients from occurring at the reactor vessel and primary heat transport outlet piping, provides instrumentation to monitor core performance, provides support for the control rod drivelines, and provides secondary holddown of the core. All of the

structural analysis aspects of assuring the UIS is structurally adequate are presented including simplified and rigorous inelastic analysis methods, elevated temperature criteria, environmental effects on material properties, design techniques, and manufacturing constraints.

- 783 (CONF-790615--13) EVALUATION OF THE INFLUENCE OF SEISMIC RESTRAINT CHARACTERISTICS ON BREEDER REACTOR PIPING SYSTEMS. Mello, R.M.; Pollono, L.P. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract EW-76-C-15-2395. 19p. Dep. NTIS, PC A02/MF A01.

From 3. US national congress on pressure vessels and piping; San Francisco, CA, USA (25 Jun 1979).

For the Clinch River Breeder Reactor Plant (CRBRP) heat transport system piping within the reactor containment building, dynamic analyses of the piping loops have been performed to study the effect of restraint stiffness on the dynamic behavior of the piping. In addition, analysis and testing of typical CRBRP restraint system components have been performed for the purpose of quantifying and verifying the basic characteristics of the restraints used in the piping system dynamic analysis.

- 784 (CONF-790615--16) DESIGN CONSIDERATIONS FOR CRBRP HEAT TRANSPORT SYSTEM PIPING OPERATING AT ELEVATED TEMPERATURES. Pollono, L.P.; Mello, R.M. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract EY-76-C-15-2395. 27p. Dep. NTIS, PC A03/MF A01.

From 3. US national congress on pressure vessels and piping; San Francisco, CA, USA (25 Jun 1979).

The heat transport system sodium piping for the Clinch River Breeder Reactor Plant (CRBRP) within the reactor containment building must withstand high temperatures for long periods of time. Each phase of the mechanical design process of the piping system is influenced by elevated temperature considerations which include material thermal creep effects, ratchetting caused by rapid temperature transients and stress relaxation, and material degradation effects. The structural design philosophy taken to design the CRBRP piping operating in a high temperature environment is described. The resulting design of the heat transport system piping is presented along with a discussion of special features that resulted from the elevated temperature considerations.

- 785 (CONF-790731--1) CHARGED PARTICLE IRRADIATIONS IN THE UNITED STATES BREEDER REACTOR PROGRAM. Rowcliffe, A.F.; Lee, E.H.; Sklad, P.S. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 24p. Dep. NTIS, PC A02/MF A01.

From US/USSR breeder reactor core materials information exchange meeting; Richland, WA, USA (Jul 1979).

Charged particle irradiations used as a means of studying radiation damage in materials of interest in the US Breeder Reactor Program are reviewed.

- 786 (CONF-790802--48) THERMAL AND THERMAL STRESS ANALYSIS OF A POOL TYPE LMFBR DECK STRUCTURE. Stephenson, M.; Ahmed, H.; Ma, D. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 17p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R.

Germany (13 Aug 1979).

The thermal capabilities of the ANSYS code were used to construct a thermal model of a pool type LMFBR system. This model included the primary tank of hot sodium covered by the concrete deck. Included in the geometry were the pump, heat exchanger, and control rod penetrations, and the metallic insulation underneath the concrete. The model included radiation and conduction of heat from the hot sodium to the concrete deck, and the cooling of the concrete via water pipes embedded in it. Thermal stresses are calculated within the code for the various plate and shell steel elements used in the code to represent different parts of the structure. This study provides a basis for studying the effect of insulation and coolant in the design of the deck structure.

- 787 (CONF-790802--49) COMPARISON OF TRANSIENT PCRV MODEL TEST RESULTS WITH ANALYSIS. Marchertas, A.H.; Belytschko, T.B. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 19p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

Comparisons are made of transient data derived from simple models of a reactor containment vessel with analytical solutions. This effort is a part of the ongoing process of development and testing of the DYNAPCON computer code. The test results used in these comparisons were obtained from scaled models of the British sodium cooled fast breeder program. The test structure is a scaled model of a cylindrically shaped reactor containment vessel made of concrete. This concrete vessel is prestressed axially by holddown bolts spanning the top and bottom slabs along the cylindrical walls, and is also prestressed circumferentially by a number of cables wrapped around the vessel. For test purposes this containment vessel is partially filled with water, which comes in direct contact with the vessel walls. The explosive charge is immersed in the pool of water and is centrally suspended from the top of the vessel. The load history was obtained from an ICECO analysis, using the equations of state for the source and the water. A detailed check of this solution was made to assure that the derived loading did provide the correct input. The DYNAPCON code was then used for the analysis of the prestressed concrete containment model. This analysis required the simulation of prestressing and the response of the model to the applied transient load. The calculations correctly predict the magnitudes of displacements of the PCRV model. In addition, the displacement time histories obtained from the calculations reproduce the general features of the experimental records: the period elongation and amplitude increase as compared to an elastic solution, and also the absence of permanent displacement. However, the period still underestimates the experiment, while the amplitude is generally somewhat large.

- 788 (CONF-790802--50) DYNAMIC RESPONSE OF CRACKED HEXAGONAL SUBASSEMBLY DUCTS. Glazik, J.L.; Petroski, H.J. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 17p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

The hexagonal subassembly ducts (hexcans) of current Liquid Metal Fast Breeder Reactor (LMFBR) designs are typically made of 20% coldworked Type 316 stainless steel. Prolonged

exposure of this initially tough and ductile material to a fast neutron flux at high temperatures can result in severe embrittlement. Under these conditions, the unstable crack propagation of flaws, which may have been introduced during fabrication or transportation of the hexcans, is a problem of interest in LMFBR safety analysis. The abnormal overpressurization resulting from certain interactions within a subassembly, or the rupture of one or more fuel pins, may be sufficient to overload an otherwise subcritical crack in an embrittled hexcan. This paper examines the dynamic elastic response of flawed and unflawed fast reactor subassembly ducts. A plane-strain finite element analysis was performed for ducts containing internal corner cracks, as well as external midflat cracks. Two worst case loading situations were considered: rapid uniform internal pressurization and suddenly applied point loads at opposite midflats. The finite-element code CHILES, which can accommodate the stress singularities that occur at crack tips, was given dynamic capabilities through the inclusion of a consistent mass matrix and step-by-step time integration scheme. The SAP IV code was also employed for eigenvalue analysis and modal response. Although this code does not contain singular elements in its element library, dynamic stress intensity factors were calculated by a technique requiring only ordinary isoparametric quadrilaterals.

- 789 (CONF-790802--51) FRACTURE EXPERIMENTS WITH CRACKED AND EMBRITTLED HEXCAN SECTIONS. Petroski, H.J.; Yamada, H. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 15p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

Crack-like flaws that might exist in hexagonal subassembly ducts are not generally expected to initiate brittle fracture, but there is reason to be concerned about the integrity of cracked hexcans that have been severely embrittled by fast-neutron irradiation. In order to check the applicability of linear elastic fracture mechanisms analysis to such ducts, a program of experiments with deliberately cracked and embrittled hexcan sections has been undertaken. Type 446 stainless steel was used for the test sections because this material may be heat treated to alter its room-temperature mechanical properties to such an extent that they simulate those of irradiated 20% cold-worked Type 316 stainless steel at reactor operating temperature. Although these different materials are not metallurgically comparable in their fracture properties, the structural response of hexcan sections fabricated from Type 446 may be assumed similar to that of irradiated Type 316 stainless steel hexcans. A series of experiments involving several different crack depths at either the corners or midflats of hexcan sections has been begun. After heat treatment and fatigue-sharpening of the cracks, the duct sections were loaded at two opposite midflats, and the load vs crack opening displacement was recorded for each test. Earlier analytical work provides stress intensity factor calibrations for the test conditions, and these calibrations enable one to predict analytically the load at which brittle fracture could occur. In this paper, a comparison of these analytical predictions and experimental results will be made and discussed.

- 790 (CONF-790802--52) THREE-DIMENSIONAL

FINITE-ELEMENT COMPUTATIONS OF THE TRANSIENT RESPONSE OF COMPONENTS TYPICAL OF A REACTOR STRUCTURE. Ash, J.E.; Kulak, R.F.; Marchertas, A.H. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 17p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

A structural analysis, finite-element code SADCAT has been developed to compute the elastoplastic response of thin-walled structural members undergoing rapid, large displacements in three-dimensional space. A convected-coordinate technique permits a simplified description of large displacements and rotations. The computational procedure consists of a lumped-mass, explicit numerical scheme suited to the nonlinear path-dependent material behavior. Code predictions have been compared with analytical calculations and published experimental data, and the correlations validate the capability of the code to provide reliable predictions for three-dimensional shell-type structure.

791 (CONF-790802--55) EVALUATION OF LAGRANGIAN, EULERIAN, AND ARBITRARY LAGRANGIAN-EULERIAN METHODS FOR FLUID-STRUCTURE INTERACTION PROBLEMS IN HCDA ANALYSIS. Chang, Y.W.; Chu, H.Y.; Gvildys, J.; Wang, C.Y. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 17p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

The analysis of fluid-structure interaction involves the calculation of both fluid transient and structure dynamics. In the structural analysis, Lagrangian meshes have been used exclusively, whereas for the fluid transient, Lagrangian, Eulerian, and arbitrary Lagrangian-Eulerian (quasi-Eulerian) meshes have been used. This paper performs an evaluation on these three types of meshes. The emphasis is placed on the applicability of the method in analyzing fluid-structure interaction problems in HCDA analysis.

792 (CONF-790802--56) COMPARISONS OF REXCO CODE PREDICTIONS WITH SRI SM-2 EXPERIMENTAL RESULTS. Chang, Y.W.; Gvildys, J. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 19p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

SRI International has performed a series of simple model (SM) experiments for the Clinch River Breeder Reactor Project (CRBR). The SM tests consisted of five experiments. The energy source used to simulate the core-disassembly accident loads was a PETN-microsphere mixture, which was well calibrated. These experiments were well instrumented and performed under carefully controlled conditions. The experimental data can be used as reliable test data for validation of computer codes, as well as the modeling technique used in the computer analysis. This paper deals with the REXCO-HEP code predictions of the SRI SM-2 test, which was a dynamic test performed on 1/20th scale model of the CRBR. Two calculations have been performed: one used the pressure history $P(t)$ of the core detonation products as input, and the other the pressure-volume relation ($P=V$) of the detonation products as input. The pressure loadings and wall deformations obtained with the $P(t)$ calculations are in better agreement with the experimental measurements than those

obtained with the $P=V$ calculations. This is because the $P=V$ relations used in the code calculation were derived from the pressure gauge readings of the core gas, the measured surface motions of the slug and the calculated compressibility of the coolant; they may become less accurate at low pressures. Therefore, in performing code validation calculations, the pressure history of the core gas should be used in the analysis to describe the behavior of the core gas, if the $P(t)$ values are available from the experimental data.

793 (CONF-790802--59) EFFECTS OF GAP SIZES ON STRESSES AND DEFLECTIONS OF A FUEL ELEMENT. Wu, T.S. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 14p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

Based on the simple beam theory and the principle of superposition, the elastic lateral deflection and flexural stress of wire-wrapped fuel elements were studied. In wire-wrapped fuel elements, the space wire provides potential contact points for neighboring fuel elements. The actual points of contact, however, depend on the deflection of the fuel element. The model used in this investigation consists of a cantilever beam with intermediate stops located at a distance g from the axis of the beam, where g is the gap size. Three values of $g = 0, 0.051 \text{ mm}$ (2 mil), and 0.127 mm (5 mil) were used in this investigation. Results from two wire-wrapped fuel elements under their respective transverse thermal gradients show that both the number of contact points and the magnitude of the maximum reaction force are reduced as the value of g increases. In all cases considered, the maximum reaction force is acting at the stop farthest from the fixed end, and the maximum deflection occurs in the vicinity where thermal gradient had its peak value. Another expected result is that the maximum lateral deflection increases with the gap size. The largest deflection for all cases studied, however, is still much smaller than the radius of a typical space wire. This suggests that, under the conditions such as the ones considered here, direct contact between a fuel element and its adjacent member is very unlikely. The maximum flexural stresses with and without gap differ only slightly, and the relation between the stress and the gap size is not monotonic. Since a larger gap will only reduce stress slightly, but will cause larger deflection and therefore more positive thermal-hydraulic feedback, a smaller gap that is able to keep the maximum stress within the design limit should be favorably considered.

794 (CONF-790802--61) COUPLED HYDRODYNAMIC-STRUCTURAL ANALYSIS OF AN INTEGRAL FLOWING SODIUM TEST LOOP IN THE TREAT REACTOR. Zeuch, W.R.; A-Moneim, M.T. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 16p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

A hydrodynamic-structural response analysis of the Mark-IIICB loop was performed for the TREAT (Transient Reactor Test Facility) test AX-1. Test AX-1 is intended to provide information concerning the potential for a vapor explosion in an advanced-fueled LMFBR. The test will be conducted in TREAT with unirradiated uranium-carbide fuel pins in the Mark-IIICB integral flowing sodium loop. Our analysis addressed the ability of the experimental hardware to maintain its containment integrity during the

reference accident postulated for the test. Based on a thermal-hydraulics analysis and assumptions for fuel-coolant interaction in the test section, a pressure pulse of 144 MPa maximum pressure and pulse width of 1.32 ms has been calculated as the reference accident. The response of the test loop to the pressure transient was obtained with the ICEPEL and STRAW codes. Modelling of the test section was completed with STRAW and the remainder of the loop was modelled by ICEPEL.

- 795 (CONF-790802--62) THERMOHYDRAULIC AND THERMAL STRESS ASPECTS OF A POROUS BLOCKAGE IN AN LMFBR FUEL ASSEMBLY. Kuzay, T.M.; Marr, W.W.; Helenberg, H.W.; Ariman, T.; Wilson, R.E.; Pedersen, D.R. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 15p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

The current safety scenarios of Liquid Metal Fast Breeder Reactors (LMFBR) under local fault propagation include the study of a hypothetical accident initiated by the formation of an external debris porous blockage in a fuel subassembly. In this preliminary experimental and analytical investigation, a non-heat-generating porous blockage was postulated to cover 18 flow channels of a 37 pin Fast Test Reactor (FTR) type fuel subassembly. The axial extent of the blockage is 50 mm. The blockage material is stainless steel (SS 316) with 30 percent average porosity (percent void volume). The blockage and the pins were modeled with a finite element technique and the thermal field in the blockage was predicted. This thermal field was utilized to do a planar thermal stress analysis of the postulated blockage. To verify the analytical model and also to better understand the thermal-hydraulics of such a porous blockage out-of-pile tests were conducted in a sodium loop. Data from the out-of-pile tests was utilized to calibrate and improve the analytical model.

- 796 (CONF-790802--63) STUDY OF STRUCTURAL ATTACHMENTS OF A POOL TYPE LMFBR VESSEL THROUGH SEISMIC ANALYSIS OF A SIMPLIFIED THREE DIMENSIONAL FINITE ELEMENT MODEL. Ahmed, H.; Ma, D. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 15p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

A simplified three dimensional finite element model of a pool type LMFBR in conjunction with the computer program ANSYS is developed and scoping results of seismic analysis are produced. Through this study various structural attachments of a pool type LMFBR like the reactor vessel skirt support, the pump support and reactor shell-support structure interfaces are studied. This study also provides some useful results on equivalent viscous damping approach and some improvements to the treatment of equivalent viscous damping are recommended. This study also sets forth pertinent guidelines for detailed three dimensional finite element seismic analysis of pool type LMFBR.

- 797 (CONF-790802--65) THREE-DIMENSIONAL FINITE ELEMENT FORMULATION FOR FLUID-STRUCTURE INTERACTION. Kulak, R.F. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 14p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R.

Germany (13 Aug 1979).

The safety evaluation of fast reactors often involves the modeling of various types of structural components which interact with surrounding fluid in three-dimensional space. In this paper, several key areas for performing three-dimensional fluid-structure interaction calculations are discussed. A new development is presented for a three-dimensional hexahedral hydrodynamic finite-element based upon a Quasi-Eulerian approach. By using trilinear shape functions and assuming a constant pressure field within the element, simple relations are obtained for internal nodal forces. The forms developed are easily coded and computationally efficient. Because the formulation is based upon a rate approach it is applicable to problems involving large displacements. Only adiabatic processes are considered. The difficult tasks (1) of input data preparation and checking and (2) the displaying of computed results are discussed. The use of automatic mesh generators, input data error analysis, and graphical display of the finite element meshes used to model reactor components are shown to ease the input task. In the final section, the formulation is applied to the three-dimensional fluid-structure interactions involved in a current reactor safety problem.

- 798 (CONF-790802--67) THREE-DIMENSIONAL FLUID-STRUCTURE INTERACTION DYNAMICS OF A POOL-REACTOR IN-TANK COMPONENT. Kulak, R.F. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 16p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

The safety evaluation of reactor-components often involves the analysis of various types of fluid/structural components interacting in three-dimensional space. For example, in the design of a pool-type reactor several vital in-tank components such as the primary pumps and the intermediate heat exchangers are contained within the primary tank. Typically, these components are suspended from the deck structure and largely submersed in the sodium pool. Because of this positioning these components are vulnerable to structural damage due to pressure wave propagation in the tank during a CDA. In order to assess the structural integrity of these components it is necessary to perform a dynamic analysis in three-dimensional space which accounts for the fluid-structure coupling. A model is developed which has many of the salient features of this fluid-structural component system.

- 799 (CONF-790802--68) STRUCTURAL FORMULATION FOR UPPER INTERNALS OF LMFBR. Kennedy, J.M.; Belytschko, T. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 16p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

The upper internal structures which are located above the core and below the head cover may play a significant role in the hydrodynamics of an energy excursion such as a hypothetical core disruptive accident (HCDA) by mitigating the slug impact on the head. For purposes of studying these effects, a structural model was developed for the upper internals and the support columns which connect them to the head of the reactor vessel. This structural model can be used either by itself or in an interactive mode with one of the hydrodynamics codes developed at Argonne, such

as ICECO or REXCO.

- 800 (CONF-790802--69) CUMULATIVE DAMAGE FRACTION DESIGN APPROACH FOR LMFBR METALLIC FUEL ELEMENTS. Johnson, D.L.; Einziger, R.E.; Huchman, G.D. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 14p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

The cumulative damage fraction (CDF) analytical technique is currently being used to analyze the performance of metallic fuel elements for proliferation-resistant LMFBRs. In this technique, the fraction of the total time to rupture of the cladding is calculated as a function of the thermal, stress, and neutronic history. Cladding breach or rupture is implied by $CDF = 1$. Cladding wastage, caused by interactions with both the fuel and sodium coolant, is assumed to be uniformly thin the cladding wall. The irradiation experience of the EBR-II Mark-II driver fuel with solution-annealed Type 316 stainless steel cladding provides an excellent data base for testing the applicability of the CDF technique to metallic fuel. The advanced metal fuels being considered for use in LMFBRs are U-15-Pu-10Zr, Th-20Pu and Th-20U (compositions are given in weight percent). The two cladding alloys being considered are Type 316 stainless steel and a titanium-stabilized Type 316 stainless steel. Both are in the cold-worked condition. The CDF technique was applied to these fuels and claddings under the assumed steady-state operating conditions.

- 801 (CONF-790802--70) FLUID-STRUCTURE INTERACTION ANALYSIS OF A DECK STRUCTURE DURING A HCDA. Kulak, R.F. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 19p. Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

Presented is an assessment of the structural integrity of the deck structure of a pool-type LMFBR during a Hypothetical Core Disruptive Accident (HCDA). During this accident the sodium above the core is propelled upward until it impacts against the deck structure. This hydrodynamic loading could produce (1) significant structural damage and (2) sodium leak paths. A finite-element model is used to study the deck dynamics during slug impact. By using the symmetry of the system, a sector model which accounts for the salient features of the system is developed. The main radial I-beam, component support I-beam and bottom annular plate are modeled using triangular plate elements. The concrete fill is modeled using hexahedral continuum elements. Using the above finite-element model the dynamics of the deck during a HCDA are investigated.

- 802 (CONF-790802--73) TRANSIENT ANALYSIS OF LMFBR REINFORCED/PRESTRESSED CONCRETE CONTAINMENT. Marchertas, A.H.; Belytschko, T.B.; Bazant, Z.P. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 16p. Dep. NTIS, PC A02.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

The use of prestressed concrete reactor vessels (PCRVs) for LMFBR containment creates a need for analytical methods for treating the transient response of such structures, for LMFBR containment must be capable of sustaining the dynamic effects which arise in a

hypothetical core disruptive accident (HCDA). These analyses require several unique features: a model of concrete which includes tensile cracking, a methodology for representing the prestressing tendons and for simulating the prestressing operation, and an efficient computational tool for treating the transient response. For the purpose of treating the transient response, a finite element program with explicit time integration was chosen. For the purpose of illustrating the applicability of these techniques and the validity of the models for concrete and the prestressing tendons, several example solutions are presented and compared with experimental results.

- 803 (CONF-790808--1) TESTING OF THE CRBRP DIRECT HEAT REMOVAL SERVICE IN A 1/21 SCALE MODEL. Novendstern, E.H.; Marinkovich, P.S. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Aug 1979. Contract EY-76-C-15-2395. 17p. Dep. NTIS, PC A02/MF A01.

From 18. ASME national heat transfer conference; San Diego, CA, USA (Aug 1979).

The Direct Heat Removal Service (DHRS) of the Clinch River Breeder Reactor Plant provides an alternative means of cooling of the core in the extremely unlikely event that both the normal and auxiliary heat paths are not available. The hot sodium is taken from the outlet plenum, cooled and returned into the outlet plenum at a different location; this sodium is mixed in the outlet plenum and pumped through the primary loops into the reactor core to remove the decay heat. The hydraulic performance of the outlet plenum was evaluated with a test using a 1/21 scale outlet plenum feature model. The model hydraulically simulated the Upper Internals Structure and the sodium within the reactor vessel. The hydraulic characteristics were studied to determine the amount of mixing which occurs in the outlet plenum.

- 804 (CONF-790808--13) COMPARISON OF NUMERICAL RESULTS WITH EXPERIMENTAL DATA FOR SINGLE-PHASE NATURAL CONVECTION IN AN EXPERIMENTAL SODIUM LOOP. Ribando, R.J. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 53p. Dep. NTIS, PC A04/MF A01.

From 18. ASME national heat transfer conference; San Diego, CA, USA (Aug 1979).

A comparison is made between computed results and experimental data for single-phase natural convection in an experimental sodium loop. The tests were conducted in the Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility, an engineering-scale high temperature sodium facility at the Oak Ridge National Laboratory used for thermal-hydraulic testing of simulated LMFBR subassemblies at normal and off-normal operating conditions. Heat generation in the 19 pin assembly during these tests was typical of decay heat levels. Tests were conducted both with zero initial forced flow and with a small initial forced flow. The bypass line was closed in most tests, but open in one. The computer code used to analyze these tests [LONAC (Low flow and Natural Convection)] is an ORNL-developed, fast running, one-dimensional, single-phase finite difference model for simulating forced and free convection transients in the THORS loop.

- 805 (CONF-790808--18) EXPERIMENTAL STUDY OF THE TRANSITION FROM FORCED TO NATURAL CIRCULATION IN EBR-II AT LOW POWER AND FLOW. Gillette, J.L.; Singer, R.M.; Tokar, J.V.;

- Sullivan, J.E. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 9p. Dep. NTIS, PC A02/MF A01.
- From 18. ASME national heat transfer conference; San Diego, CA, USA (Aug 1979).
- A series of tests have been conducted in EBR-II which studied the dynamics of the transition from forced to natural circulation flow in a liquid-metal-cooled fast breeder reactor (LMFBR). Each test was initiated by abruptly tripping an electromagnetic pump which supplies 5 to 6% of the normal full operational primary flow rate. The ensuing flow coast-down reached a minimum value after which the flow increased as natural circulation was established. The effects of secondary system flow through the intermediate heat exchanger and reactor decay power level on the minimum in-core flow rates and maximum in-core temperatures were examined.
- 806 (CONF-790808--19) TEMPERATURES OF EBR-II SUBASSEMBLIES IN AIR OR ARGON WITHOUT FORCED COOLING. Koenig, J.F.; Buchanan, T.N. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 12p. Dep. NTIS, PC A02/MF A01.
- From 18. ASME national heat transfer conference; San Diego, CA, USA (Aug 1979).
- Test were conducted to determine the maximum temperatures in a electrically heated tube bundle not subject to forced cooling. While natural convection cooling through the bundle was allowed, it was found that the majority of the heat was removed by radial conduction and radiation to the surrounding hex container. Overall heat transfer coefficients are reported for the combined heat removal.
- 807 (CONF-790816--2) OPTIMIZATION AND COMMON CAUSE ANALYSIS OF A PROTOTYPE LARGE BREEDER REACTOR SAFETY SYSTEM. Rasmuson, D.M.; Wilson, J.R.; Burdick, G.R. (Idaho National Engineering Lab., Idaho Falls (USA)). 1979. Contract EY-76-C-07-1570. 5p. Dep. NTIS, PC A02/MF A01.
- From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).
- The prototype large breeder reactor (PLBR) is a 1000 MWe sodium cooled fast breeder reactor designed to succeed the Clinch Breeder Reactor and verify scaling laws. Several conceptual PLBR designs have been presented for consideration. A procedure is needed for the conceptualization of, evaluation of, and selection between such designs. This paper presents a method that can be effectively used to: (1) allocate optimal component redundancy, (2) evaluate design trade-offs, and (3) help in design finalization. The procedure is applied to three diverse design options for a shutdown heat removal system (SHRS). The selected configuration is analyzed for potential common cause failures.
- 808 (CONF-790816--12) DUAL CHANGE-OF-PHASE IN FORCED FLOW ON A FINITE WALL, WITH APPLICATION TO SLSF SAFETY ANALYSIS. El-Genk, M.S.; Moore, R.L. (Idaho National Engineering Lab., Idaho Falls (USA)). 1979. Contract EY-76-C-07-1570. 11p. Dep. NTIS, PC A02/MF A01.
- From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).
- An important study related to the safety analysis of Liquid Metal Fast Breeder Reactors (LMFBRs) is to understand the phenomena of molten fuel relocation and refreezing on the core cold structures following a hypothesized loss-of-flow accident (LOFA). An assessment of molten fuel containment capability for the Sodium Loop Safety Facility (SLSF) in-pile experiments following a postulated LOFA is presented, considering the conditions of molten fuel flow, simultaneous melting in the outer duct wall, and convective cooling at the outer surface of the wall due to sodium coolant bypass flow.
- 809 (CONF-790816--18) BEHAVIOR OF SODIUM--OXIDE AND URANIUM--OXIDE AEROSOLS IN A LARGE VESSEL. Adams, R.E.; Kress, T.S.; Han, J.T.; Parsly, L.F. Jr. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 6p. Dep. NTIS, PC A02/MF A01.
- From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).
- Results are reported for experiments on the behavior of sodium--oxide aerosols, uranium--oxide aerosols, and various mixtures of these in the Nuclear Safety Pilot Plant (NSPP) vessel at Oak Ridge. The NSPP Program, sponsored by the Nuclear Regulatory Commission, is to provide experimental qualification of the HAARM-3 aerosol behavioral code being independently developed at Battelle-Columbus Laboratories. The results from ten single-component and two mixed component aerosol experiments are presented along with selected comparisons with the code predictions.
- 810 (CONF-790816--19) ANALYSIS OF SODIUM VALVE RELIABILITY DATA AT CREDO. Bott, T.F.; Haas, P.M. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 13p. Dep. NTIS, PC A02/MF A01.
- From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).
- The Centralized Reliability Data Organization (CREDO) has been established at Oak Ridge National Laboratory (ORNL) by the Department of Energy to provide a centralized source of data for reliability/maintainability analysis of advanced reactor systems. The current schedule calls for development of the data system at a moderate pace, with the first major distribution of data in late FY-1980. Continuous long-term collection of engineering, operating, and event data has been initiated at EBR-II and FFTF.
- 811 (CONF-790816--20) NATURAL CONVECTION BOILING OF SODIUM IN A SIMULATED FBR FUEL ASSEMBLY SUBCHANNEL. Garrison, P.W.; Morris, R.H.; Montgomery, B.H. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 12p. Dep. NTIS, PC A02/MF A01.
- From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).
- A series of single channel sodium boiling experiments were conducted in the Sodium Boiling Test (SBT) Facility at the Oak Ridge National Laboratory. The objective of these tests was to determine the maximum power that can be transferred to the coolant in a fuel assembly subchannel when the flow is driven by natural convection. Stable boiling was achieved for test times approaching one hour in duration for powers below the critical level. The power corresponding to dryout in these tests compares well with predictions from an earlier water simulation of fuel assembly voiding behavior.
- 812 (CONF-790816--21) EFFECTS OF SCATTERING ON RADIATIVE COOLING OF A LARGE HCDA-PRODUCED VAPOR BUBBLE. Oezisik, M.N.; White, J.S.; Kress, T.S. (Oak Ridge National Lab., TN

(USA)). 1979. Contract W-7405-ENG-26. 9p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Previous investigations have shown that radiation is potentially a significant mode of energy transfer for the cooling of a large HCDA-produced vapor bubble. The radiative cooling capability could be significantly reduced by the presence of small particles or droplets due to local condensation or fog formation. The purpose of this work is to demonstrate the separate effect of scattering on the radiative heat flux in order to determine the relative importance to conductive and convective heat transfer.

- 813 (CONF-790816--22) EVALUATION OF COBRA III-C AND SABRE-I (WIRE WRAP VERSION) COMPUTATIONAL RESULTS BY COMPARISON WITH STEADY-STATE DATA FROM A 19-PIN INTERNALLY GUARD HEATED SODIUM COOLED BUNDLE WITH A SIX-CHANNEL CENTRAL BLOCKAGE (THORS BUNDLE 3C). Dearing, J.F.; Rose, S.D.; Nelson, W.R. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 17p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The predicted computational results of two well-known sub-channel analysis codes, COBRA-III-C and SABRE-I (wire wrap version), have been evaluated by comparison with steady state temperature data from the THORS Facility at ORNL. Both codes give good predictions of transverse and axial temperatures when compared with wire wrap thermocouple data. The crossflow velocity profiles predicted by these codes are similar which is encouraging since the wire wrap models are based on different assumptions.

- 814 (CONF-790816--24) THORS: A HIGH-TEMPERATURE SODIUM TEST FACILITY RATED AT 2.0 MW. Gnadt, P.A.; Anderson, A.H.; Clapp, N.E.; Montgomery, B.H.; Collins, C.W.; Stulting, R.D. (Oak Ridge National Lab., TN (USA); Union Carbide Corp., Oak Ridge, TN (USA). Nuclear Div.). 1979. Contract W-7405-ENG-26. 12p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The Thermal-Hydraulic Out-of-Reactor Safety (THORS) facility at Oak Ridge National Laboratory (ORNL) is a high-temperature sodium test facility operated for the United States Breeder Reactor Safety Program. The facility is primarily used for testing large simulated Liquid-Metal Fast Breeder Reactor (LMFBR) fuel subassemblies. The facility has recently been upgraded to provide a 2.0-MW test bundle power input and heat removal capability. A new test section, which will be capable of operating at 980°C and which will accommodate a 217-pin bundle, has also been added. A 61-pin bundle is currently under test in the facility. A description of the test facility is presented, along with a brief summary of the 8-year operating history of this safety-related test facility.

- 815 (CONF-790816--28) DESIGN OPTIMIZATION OF A PROTOTYPE LARGE BREEDER REACTOR SAFETY SYSTEM. Rasmuson, D.M.; Wilson, J.R.; Burdick, G.R. (Idaho National Engineering Lab., Idaho Falls (USA)). 1979. Contract EY-76-C-07-1579. 12p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug

1979).

A procedure is presented that can be used in the conceptualization of, evaluation of, and selection among several conceptual liquid metal fast breeder reactor (LMFBR) designs. The method can be effectively used to: (a) allocate optimal component redundancy, (b) evaluate design trade-offs, and (c) help in design finalization. The procedure is applied to three diverse design options for a shutdown heat removal system. The selected configuration is analyzed for potential common cause failures.

- 816 (CONF-790816--29) APPLICATION OF GENERALIZED PERTURBATION THEORY TO FAST REACTOR SAFETY SENSITIVITY STUDIES. Parks, C.V.; Tomlinson, E.T.; Oblo, E.M. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 11p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

A new sensitivity analysis method based on generalized perturbation theory is applied to a typical fast reactor safety code. The adjoint equations are provided for the coupled, nonlinear, conduction-convection equations as they are implemented in MELT-IIIA. An expression for the sensitivity of any temperature to problem input data is presented. In this initial application of the new methodology, two simple transient problems for the Fast Flux Test Facility are considered. Sensitivity results for various temperature responses to all thermal input data are presented. Conclusions are drawn about the practicality and advantages of developing this new sensitivity methodology. Comparisons of perturbation theory with response surface methodologies are also made throughout the paper.

- 817 (CONF-790816--30) ANALYSIS OF THE HEAT AND MASS TRANSFER PROCESSES OF UO_2 BUBBLE IN SODIUM. Tobias, M.L. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 6p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The anticipated behavior of uranium oxide vapor bubbles produced by the capacitor discharge vaporization (CDV) method in the Fuel Aerosol Simulant Test (FAST) Facility is discussed on the basis of relatively simple physical models. Results of calculations for the rate of bubble rise and for heat and mass transfer rates are presented. Parametric studies indicate that future analysis efforts should emphasize the diffusion condensation process and the loss of heat from the bubble by radiation. Transfer of heat in the surrounding sodium is rapid enough that simplified models should be adequate. No important effects were noted in connection with bubble depth, initial quantity of UO_2 , or initial superheat.

- 818 (CONF-790816--31) SAS3A ANALYSIS OF NATURAL CONVECTION BOILING BEHAVIOR IN THE SODIUM BOILING TEST FACILITY. Klein, G.A. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 8p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

An analysis of natural convection boiling behavior in the Sodium Boiling Test (SBT) Facility has been performed using the SAS3A computer code. The predictions from this analysis indicate that stable boiling can be

achieved for extensive periods of time for channel powers less than 1.4 kW and indicate intermittent dryout at higher powers up to at least 1.7 kW. The results of this analysis are in reasonable agreement with the SBT Facility test results.

- 819 (CONF-790816--32) SODIUM BOILING INCOHERENCE IN A 19-PIN WIRE-WRAPPED BUNDLE. Wantland, J.L.; Garrison, P.W.; Nelson, W.R.; Dearing, J.F.; Gnadt, P.A.; Ribando, R.J.; Fontana, M.H. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 9p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

During various flow transients in a sodium-cooled reactor, localized boiling can occur. If this boiling does not result in dryout, significant reactor core damage is not likely. A full-length electrically heated 19-pin bundle was used to determine the extent to which dynamic boiling can be sustained before dryout occurs. Over 30 boiling runs were made with runs at three flow-power conditions culminating in dryout. Continuous boiling for time periods exceeding 20 sec was observed. Preliminary data analysis suggested that thermal inertia of the duct walls, which were backed with thermal insulation, was higher than designed and was contributing to boiling incoherence. Posttest examination confirmed that the insulation annulus had become permeated with sodium, resulting in a significantly increased thermal inertia. Detailed comparisons of experimental results with the results of several different analytical techniques indicate that incoherent boiling caused by bundle thermal inertia was responsible for the long time periods between boiling inception and dryout. This suggests that thermal inertia designed into the reactor core could prevent or delay core damage during various flow-power mismatch transients.

- 820 (CONF-790816--37) CRBRP STRUCTURAL AND THERMAL MARGIN BEYOND THE DESIGN BASE. Strawbridge, L.E. (Westinghouse Electric Corp., Madison, PA (USA)). Advanced Reactors Div.). 1979. Contract EY-76-C-15-2395. 13p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Although Hypothetical Core Disruptive Accidents (HCDAs) are not Design Basis Accidents for the Clinch River Breeder Reactor Plant, extensive assessments of HCDA consequences have been made. Prudent margins beyond the design base have been included in the design to further reduce the risk to the public from highly improbable occurrences. These margins include Structural Margin Beyond the Design Base to address the energetics aspects and Thermal Margin Beyond the Design Base to address the longer term thermal and radiological consequences. The assessments that led to the specification of these margins are described, along with the experimental support for those assessments.

- 821 (CONF-790816--39) COMPARISON OF COBRA III-C AND SABRE-1 (WIRE-WRAP VERSION) COMPUTATIONAL RESULTS WITH STEADY-STATE DATA FROM A 19-PIN INTERNALLY GUARD HEATED SODIUM-COOLED BUNDLE WITH A SIX-CHANNEL CENTRAL BLOCKAGE (THORS BUNDLE 3C). Dearing, J.F.; Nelson, W.R.; Rose, S.D. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 11p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor

safety technology; Seattle, WA, USA (19 Aug 1979).

Computational thermal-hydraulic models of a 19-pin, electrically heated, wire-wrap liquid-metal fast breeder reactor test bundle were developed using two well-known subchannel analysis codes, COBRA III-C and SABRE-1 (wire-wrap version). These two codes use similar subchannel control volumes for the finite difference conservation equations but vary markedly in solution strategy and modeling capability. In particular, the empirical wire-wrap-forced diversion crossflow models are different. Surprisingly, however, crossflow velocity predictions of the two codes are very similar. Both codes show generally good agreement with experimental temperature data from a test in which a large radial temperature gradient was imposed. Differences between data and code results are probably caused by experimental pin bowing, which is presently the limiting factor in validating coded empirical models.

- 822 (CONF-790816--55) OPERATIONAL- SAFETY- TESTING EXPERIENCE AT EBR-II. Sackett, J.I.; Dean, E.M.; Fryer, R.M.; Golden, G.H.; Lambert, J.D.B.; Lehto, W.K.; Singer, R.M. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 10p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Operation of EBR-II has required that issues of safety and reliability be continually addressed. This has been necessary not only to support routine operation, but also to support test programs that bear upon safety. Major issues are: (1) operability of fuel elements with breached cladding (local fault prevention); (2) assurance of adequate convective cooling upon loss of pumping power (decay heat removal); and (3) demonstration of benign response of fuel-elements and subassemblies to protected overpower transients (shutdown system margin and reliability). Test programs to address these issues at EBR-II are described. Particular emphasis is given to the potential capability at EBR-II for mild transient testing, consistent with its role in addressing LOA-1 (line of assurance one) safety issues.

- 823 (CONF-790816--57) STUDIES RELATED TO EMERGENCY DECAY HEAT REMOVAL IN EBR-II. Singer, R.M.; Gillette, J.L.; Mohr, D.; Tokar, J.V.; Sullivan, J.E.; Dean, E.M. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 10p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Experimental and analytical studies related to emergency decay heat removal by natural circulation in the EBR-II heat transport circuits are described. Three general categories of natural circulation plant transients are discussed and the resultant reactor flow and temperature response to these events are presented. These categories include the following: (1) loss of forced flow from decay power and low initial flow rates; (2) reactor scram with a delayed loss of forced flow; and (3) loss of forced flow with a plant protective system activated scram. In all cases, the transition from forced to natural convective flow was smooth and the peak in-core temperature rises were small to moderate. Comparisons between experimental measurements in EBR-II and analytical predictions of the NATDEMO code are included.

- 824 (CONF-790816--58) EBR-II BREACHED FUEL TEST FACILITY. Lehto, W.K.; Koenig, J.F.; Seim, O.S.; Olp, R.H.; Strain, R.V.; Colburn, R.P. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 13p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The Breached Fuel Test Facility (BFTF) is a multipurpose experimental facility, designed to provide the capability to conduct and monitor safety and fuel behavior experiments under more severe conditions than previously allowed in EBR-II. The facility consists of an outer thimble assembly with an internal instrument stalk which extends from the reactor floor through the primary tank cover to the top of the core. Coolant from a breached element test is directed upward to the instrument train above the core. The BFTF has the capability to measure flow, temperature, particle size distribution and deposition, and delayed neutron levels for breach site characterization. This paper describes the design, the instrumentation, the operational safety concerns and the initial experiments.

- 825 (CONF-790816--59) MOLTEN CORE DEBRIS-SODIUM INTERACTIONS: M-SERIES EXPERIMENTS. Sowa, E.S.; Gabor, J.D.; Pavlik, J.R.; Cassulo, J.C.; Cook, C.J.; Baker, L. Jr. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 11p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Five new kilogram-scale experiments have been carried out. Four of the experiments simulated the situation where molten core debris flows from a breached reactor vessel into a dry reactor cavity and is followed by a flow of sodium (Ex-vessel case) and one experiment simulated the flow of core debris into an existing pool of sodium (In-vessel case). The core debris was closely simulated by a thermite reaction which produced a molten mixture of UO_2 , ZrO_2 , and stainless steel. There was efficient fragmentation of the debris in all experiments with no explosive interactions observed.

- 826 (CONF-790816--60) SAFETY APPROACH TO THE SELECTION OF DESIGN CRITERIA FOR THE CRBRP REACTOR REFUELING SYSTEM. Meisl, C.J.; Berg, G.E.; Sharkey, N.F. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). 1979. Contract EY-76-C-03-0824-027. 12p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The selection of safety design criteria for Liquid Metal Fast Breeder Reactor (LMFBR) refueling systems required the extrapolation of regulations and guidelines intended for Light Water Reactor refueling systems and was encumbered by the lack of benefit from a commercially licensed predecessor other than Fermi. The overall approach and underlying logic are described for developing safety design criteria for the reactor refueling system (RRS) of the Clinch River Breeder Reactor Plant (CRBRP). The complete selection process used to establish the criteria is presented, from the definition of safety functions to the finalization of safety design criteria in the appropriate documents. The process steps are illustrated by examples.

- 827 (CONF-790816--61) THERMAL INTERACTIONS OF A MOLTEN CORE DEBRIS POOL WITH SURROUNDING STRUCTURAL MATERIALS. Baker, L. Jr.; Cheung, F.B.; Farhadieh, R.; Stein, R.P.; Gabor, J.D.; Bingle, J.D. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 12p. Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Analytical and experimental results on individual aspects of the overall problem of the interaction of a large mass of LMFBR core debris with concrete or other materials are reviewed. Results of recent heat transfer experiments with molten UO_2 have indicated the importance of internal thermal radiation and methods to take account of this are developed. Effects of gas release and density difference are considered. The GROWS-2 Code is used to illustrate the effects of various assumptions.

- 828 (CONF-790884--2) DEVELOPMENT OF LIFE4-CN: A COMBINED CODE FOR STEADY-STATE AND TRANSIENT ANALYSES OF ADVANCED LMFBR FUELS. Liu, Y.Y.; Zawadzki, S.; Billone, M.C.; Nayak, U.P.; Roth, T. (Argonne National Lab., IL (USA); Northwestern Univ., Evanston, IL (USA); Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract W-31-109-ENG-38. 28p. Dep. NTIS, PC A02/MF A01.

From Post-SMIRT5 second international seminar on mathematical/mechanical modeling of reactor fuel elements; Berlin, F.R. Germany (26 Aug 1979).

The methodology used to develop the LMFBR carbide/nitride fuels code, LIFE4-CN, is described in detail along with some subtleties encountered in code development. Fuel primary and steady-state thermal creep have been used as an example to illustrate the need for physical modeling and the need to recognize the importance of the materials characteristics. A self-consistent strategy for LIFE4-CN verification against irradiation data has been outlined with emphasis on the establishment of the gross uncertainty bands. These gross uncertainty bands can be used as an objective measure to gauge the overall success of the code predictions. Preliminary code predictions for sample steady-state and transient cases are given.

- 829 (CONF-790902--3) IMPACT OF ADVANCED SYSTEMS ON LMFBR ACCIDENT ANALYSIS CODE DEVELOPMENT. Dunn, F.E.; Kyser, J.M. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 12p. Dep. NTIS, PC A02/MF A01.

From Impact of advanced systems on scientific computations; Livermore, CA, USA (12 Sep 1979).

In order to investigate the ability of an advanced computer, using currently available software, to handle large LMFBR accident analysis codes, the SAS3D code has been run on the NCAR CRAY-1. SAS3D is a large code (56,000 Fortran cards) using many different physical models and numerical algorithms, no one of which dominates the computing time. Even though SAS3D was developed on IBM computers, remarkably little effort was required to run it on the CRAY-1. Making limited use of the CRAY-1 vector capabilities, it runs a factor of 2.5 to 4 times faster on the NCAR CRAY-1 than on the ANL IBM 370-195. With minor modifications, an additional 20 to 30% speed improvement on the CRAY-1 is achieved. In the current process of completely re-writing SAS3D to make SAS4A, much

of the coding is being vectorized for the CRAY=1 without sacrificing IBM, CDC 7600, or UNIVAC performance and portability. An initial SAS4A test case runs a factor of 7.1 faster on the CRAY-1 than on the IBM 370-195.

- 830 (CONF-790905--1) RE-EVALUATING YOUR NUCLEAR PROGRAM NEEDS: HOW TO BENEFIT FROM YOUR VENDOR'S Q.A. PROGRAM. Cocoros, A.E. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract EY-76-C-15-2395. 22p. Dep. NTIS, PC A02/MF A01.

From American Society for Quality Control conference; Las Vegas, NV, USA (24 Sep 1979).

The quality assurance component control and verification program to be presented provides a cost effective approach to monitoring and controlling the implementation of the design, fabrication, inspection and shipping plans of a supplier. It attempts to coordinate and integrate quality control and verification effort of a supplier with the control and verification effort of the purchaser to obtain a composite which accomplishes a total need. Based on the competency and capabilities of the supplier the purchaser can either maximize the effort the supplier performs or he must maximize his effort to obtain an optimum mix. The ultimate goal is to utilize the supplier's quality assurance program to the greatest benefit in assuring maximum quality.

- 831 (CONF-790934--1) ANALYTICAL MODELING OF CORE HYDRAULICS AND FLOW MANAGEMENT IN BREEDER REACTORS. Carelli, M.D.; Willis, J.M. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract EY-76-C-15-2395. 7p. Dep. NTIS, PC A02/MF A01.

From 18. congress of International Association for Hydraulic Research; Cagliari, Italy (10 Sep 1979).

An analytical model representing the hydraulic behavior of the primary system of fast breeder nuclear reactors is discussed. A computer code capable of detailing the core flow distribution and characterizing the flow and pressure drop in each reactor component is presented. Application of this method to the reactor core thermal-hydraulic design has allowed optimization of the flow management with consequent upgrading in performance, reduction of unnecessary conservatism and very substantial cost savings. Typical quantitative examples are presented.

- 832 (CONF-791008--5) ULTRASONIC DYNAMIC IMAGING (UDI) SYSTEM TO DETERMINE THE SIZE AND VELOCITY OF LARGE SPHERICAL SIMULATED BUBBLES IN LIQUIDS. Lattimer, J.; Fitting, D.W.; Adler, L. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 4p. Dep. NTIS, PC A02/MF A01.

From ASNT fall conference; St Louis, MO, USA (15 Oct 1979).

As part of the nuclear safety program it is desirable to fully characterize the dynamics of a vapor bubble formed in liquid sodium. The bubble would be formed during the hypothetical case of partial core vaporization in a liquid-metal reactor. A real-time ultrasonic imaging system was developed which provides simultaneous measurement of the rise velocity and radius of large spherical bubbles. The range of applicability of the UDI system has been determined in model experiments.

- 833 (CONF-791051--16) ANALYSIS OF EBR-II BLANKET DOSIMETRY. Meneghetti, D.; Ebersole,

E.R.; Kucera, D.A.; Rempert, R.H. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 11p. Dep. NTIS, PC A02/MF A01.

From 3. ASTM-EURATOM symposium on reactor dosimetry; Ispra, Italy (1 Oct 1979).

Comparisons of reaction-rate calculations with absolute dosimetry measurements in the EBR-II blanket are made for $^{235}\text{U}/\text{sub f/}$, $^{239}\text{Pu}/\text{sub f/}$, and $^{238}\text{U}/\text{sub c/}$ reactions.

- 834 (CONF-791058--26) CROSS SECTION ADJUSTMENT APPLIED TO ESTIMATION OF UNCERTAINTY IN THE BREEDING RATIO OF AN LMFB. Marable, J.H.; Weisbin, C.R.; de Saussure, G. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 5p. Dep. NTIS, PC A02/MF A01.

From International conference on nuclear cross sections for technology; Knoxville, TN, USA (22 Oct 1979).

Eleven fast-reactor benchmark experiments and two neutron-field benchmark experiments were applied to a least-squares adjustment of a 26-group cross section library based primarily on ENDF/B-IV. The covariance data include correlations between cross sections for different energies, reactions, and materials, and between integral experiments, and covariances of calculational bias factors due to specific modeling and calculational procedures. The results of the adjustment are applied to the determination of the uncertainties in the multiplication factor and in the breeding ratio of a large LMFB design model fixed by the Large Core Code Evaluation Working Group (LCCEWG).

- 835 (CONF-791058--34) VITAMIN E: A MULTIPURPOSE ENDF/B-V COUPLED NEUTRON-GAMMA CROSS SECTION LIBRARY. Barhen, J.; Cacuci, D.G.; Ford, W.E. III; Roussin, R.W.; Wagschal, J.J.; Weisbin, C.R.; White, J.E.; Wright, R.Q. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 5p. Dep. NTIS, PC A02/MF A01.

From International conference on nuclear cross sections for technology; Knoxville, TN, USA (22 Oct 1979).

The US Department of Energy Office of Fusion Energy and the Division of Reactor Research and Technology jointly sponsored the development of a coupled fine-group cross section library (VITAMIN-C). The experience gained in the generation, validation, and utilization of the VITAMIN-C library along with its broad range of applicability has led to the request for updating this data set using ENDF/B-V. Additional support in this regard has been provided by the Defense Nuclear Agency (DNA) and by EPRI in support of weapons analyses and light water reactor shielding and dosimetry problems, respectively. The rationale for developing the multipurpose ENDF/B-V-based VITAMIN-E library is presented, with special emphasis on new models used in the data generation algorithms. The library specifications and testing procedures are also discussed in detail. The distribution of the VITAMIN-E library is currently subject to the same restrictions as the distribution of the ENDF/B-V data. 2 tables.

- 836 (CONF-791103--7(Summ)) METHODOLOGY FOR STATISTICAL DETERMINATION OF FUEL INTRA-ASSEMBLY FLOW MIXING UNCERTAINTY. Tang, Y.S.; Friedland, A.J.; Markley, R.A. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract EY-76-C-15-2395. 6p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society meeting; San

Francisco, CA, USA (12 Nov 1979).

To account for calculational and experimental uncertainties in reactor thermal-hydraulic analyses, hot channel/spot factors are usually used. This paper presents the methodology used to determine calculational uncertainty in the prediction of flow distribution between subchannels of LMFBR core assemblies. The flow distribution uncertainty may be caused by fabrication tolerance, distorted elements or inter-channel mixing.

- 837 (CONF-791103--25) MOLTEN FUEL CONTAINMENT ANALYSIS FOR SLSF EXPERIMENTS. El Genk, M.S.; Moore, R.L. (Idaho National Engineering Lab., Idaho Falls (USA)). 1979. Contract EY-76-C-07-1570. 5p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

The Sodium Loop Safety Facility (SLSF) experiments are being conducted in the Engineering Test Reactor (ETR) at the Idaho National Engineering Laboratory to furnish information concerning the behavior of fast reactor fuel elements under a hypothesized loss-of-flow accident conditions. These experiments involve measurements and observations that will permit an assessment of the location and time of sodium void initiation, the nature of sodium expulsion and reentry, the time and nature of cladding failure and motion, the molten fuel motion, and the extent of flow plugging by molten fuel freezing at the axial extremities of the subassembly. Because such high energy experiments are being conducted in-pile within the ETR facility, extensive pretest analysis was required to ensure loop integrity. One aspect of this analysis, which is the subject of this paper, was the assessment of molten fuel relocation and refreezing on the wall of the SLSF.

- 838 (CONF-791103--92) SAS3D ANALYSIS OF NATURAL CONVECTION BOILING BEHAVIOR IN THE SODIUM BOILING TEST FACILITY. Klein, G.; Dunn, F. (Oak Ridge National Lab., TN (USA); Argonne National Lab., IL (USA)). 1979. Contract W-7405-ENG-26. 6p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

The objective of the initial phase of testing in the Sodium Boiling Test (SBT) Facility, at the Oak Ridge National Laboratory, was to determine the maximum power that could be transferred by a simulated breeder reactor coolant subchannel when the coolant flow is driven by natural convection. In order to aid in the evaluation of the experimental data and to help understand the flow regimes present at the various power levels examined during this test program, a SAS3D computer model of the SBT Facility was developed.

- 839 (CONF-791103--99) DEVELOPMENT OF THE CENTRALIZED RELIABILITY DATA ORGANIZATION (CREDO). Haas, P.M.; Bott, T.F.; Knee, H.E.; Manning, J.J.; Hudson, S.D.; Greene, N.M.; Woodside, M.A. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 7p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

CREDO has been established to meet the needs of the US Breeder Reactor Program and the advanced reactor community for a centralized source of accurate reliability/maintainability data and data-related services. The center provides for a comprehensive program of collection, evaluation, and dissemination of

data and for necessary user services. A steering committee of nationally recognized experts has been formed to guide CREDO development and operation. Historic data have been collected, analyzed, and published. Routine, continuous data collection and processing has been initiated at US sites. Interfaces with existing data bases have been established. Special user services are under development.

- 840 (CONF-791103--105) EFFECTIVENESS OF THREE REBALANCE METHODS IN DEEP PENETRATION PROBLEMS. Rhoades, W.A.; Tomlinson, E.T. (Oak Ridge National Lab., TN (USA); Union Carbide Corp., Oak Ridge, TN (USA); Nuclear Div.). 1979. Contract W-7405-ENG-26. 22p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

The space-rebalance method brought important new capability to discrete-ordinates calculations by using an easily-solved procedure. Later, diffusion synthetic acceleration used the diffusion equation as the auxiliary equation. Application of this new approach has been slow, partially due to concomitant restrictions. Aull et al showed a reformulation which brought compatibility with weighted-difference and negative-fixup schemes. Miller showed that the differential forms of the conventional space rebalance (CRB) and the newer methods could be derived from a common framework. This paper will, similarly, cast the difference form of Aull's consistent diffusion acceleration (CDA) into the same form as CRB and will indicate a third approach suggested by this comparison. Variations of these methods will be tested on deep-penetration problems.

- 841 (CONF-791108--21) MODEL TO SIMULATE THE FISSION-PRODUCT TRANSPORT PROCESS IN THE EXPERIMENTAL BREEDER REACTOR II. So, B.Y.C. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 11p. Dep. NTIS, PC A02/MF A01.

From 72. AICHE meeting; San Francisco, CA, USA (25 Nov 1979).

When fission products are released from a cladding breach in EBR-II, they mix turbulently with the sodium in the core, in the upper plenum and in the intermediate heat exchanger. Eventually the fission products are discharged 12 to 13 s later into the primary tank. Fission gases migrate upward through a 9-ft layer of sodium and enter the cover gas. Loss of fission gas is due to decay, leakage of cover gas, cold trapping of iodine and bromine parents. Depending on the reactor operation requirement, it may purge with fresh argon. The assumptions made and differential equations used to develop a model for such transport are presented.

- 842 (CONF-791117--3) SYSTEM FOR NONDESTRUCTIVE ASSAY OF SPENT FUEL SUBASSEMBLIES: COMPARISON OF CALCULATIONS AND MEASUREMENTS. Ragan, G.L.; Ricker, C.W.; Chiles, M.M.; Ingersoll, D.T.; Slaughter, G.G.; Williams, L.R. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 13p. Dep. NTIS, PC A02/MF A01.

From ANS topical conference; Kiawah Island, SC, USA (26 Nov 1979).

A nondestructive assay system was developed for determining the total fissile content of spent fuel subassemblies at the head end of a reprocessing plant. The system can perform an assay in 20 min with an uncertainty of <5%. Antimony-beryllium neutrons interrogate the subassemblies, and proton recoil counters detect the resulting fission neutrons. Pulse-

height discrimination differentiates between the low-energy interrogation neutrons and the higher-energy fission neutrons. Calculated and measured results were compared for (1) interrogation-neutron penetrability, (2) fission-neutron detectability, (3) radial variation of assay sensitivity, (4) axial variation of assay sensitivity, and (5) the variation of detector count rate as a function of the number of fuel rods in a special 61-rod, LMFBR-type subassembly.

- 843 (CONF-791204--29) STUDIES RELATED TO PROTECTED LOSS OF FLOW ACCIDENTS IN A FAST BREEDER REACTOR. Singer, R.M.; Gillette, J.L.; Mohr, D. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 12p. Dep. NTIS, PC A02/MF A01.

From 2. Miami international conference on alternative energy sources; Miami Beach, FL, USA (10 Dec 1979).

Intrinsic to the design of liquid-metal-cooled fast-breeder-reactors (LMFBR) are a number of physical processes which inherently contribute to the safe operation and shutdown of the plant. Natural circulation cooling of the plant following a complete loss of all on-site electrical power is analyzed. The calculations which are presented demonstrate that the reactor core is adequately and stably cooled by natural circulation for a wide range of postulated initial operating conditions and plant design parameters.

- 844 (CONF-791204--34) CONCEPT EVALUATION OF NUCLEAR FUSION DRIVEN SYMBIOTIC ENERGY SYSTEMS. Renier, J.P.; Hoffman, T.J. (Oak Ridge National Lab., TN (USA)). 1979. Contract W-7405-ENG-26. 32p. Dep. NTIS, PC A03/MF A01.

From 2. Miami international conference on alternative energy sources; Miami Beach, FL, USA (10 Dec 1979).

This paper analyzes systems based on D-T and semi-catalyzed D-D fusion-powered U233 breeders. Two different blanket types were used: metallic thorium pebble-bed blankets with a batch reprocessing mode and a molten salt blanket with on-line continuous or batch reprocessing. All fusion-driven blankets are assumed to have spherical geometries, with a 85% closure. Neutronics depletion calculations were performed with a revised version of the discrete ordinates code XSDRN-PM, using multigroup (100 neutron, 21 gamma-ray groups) coupled cross-section libraries. These neutronics calculations are coupled with a scenario optimization and cost analysis code. Also, the fusion burn was shaped so as to keep the blanket maximum power density below a preset value, and to improve the performance of the fusion-driven systems. The fusion-driven symbiotes are compared with LMFBR-driven energy systems. The nuclear fission breeders that were used as drivers have parameters characteristic of heterogeneous, oxide LMFBRs. They are net plutonium users - the plutonium is obtained from the discharges of LWRs - and U233 is bred in the fission breeder thorium blankets. The analyses of the symbiotic energy systems were performed at equilibrium, at maximum rate of grid expansion, and for a given nuclear power demand.

- 845 (CONF-791205--2) EXPERIMENTAL STUDIES OF FLOW DISTRIBUTION IN A WIRE WRAPPED LMFBR BLANKET ASSEMBLY. Patch, L.; Rojdt, R.M.; Carelli, M.D.; Markley, R.A. (Westinghouse Research and Development Center, Pittsburgh, PA (USA); Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract EY-76-C-15-2395. 39p. Dep. NTIS,

PC A03/MF A01.

From ASME winter annual meeting; New York, NY, USA (2 Dec 1979).

An experimental investigation of the flow characteristics in a model of an LMFBR wire wrapped blanket assembly is presented. A 5:1 scale model of a Clinch River Breeder Reactor Plant blanket assembly was tested in an unheated airflow facility under Reynolds number simulation conditions. Pressure gradients, subchannel gap crossflows and local axial velocities were measured in the interior, side and corner subchannels, thus providing a complete, self-consistent set of data. The bulk of the experiments were performed for a Reynolds number of 22,000, representative of blanket full flow conditions. Additional data were calculated at intermediate ($Re = 11,800$) and low ($Re = 4,300$) flow conditions. Very detailed mappings of the axial velocity distribution in the subchannels and of the crossflow velocities in representative gaps were performed.

- 846 (CONF-791205--3) TECHNIQUES FOR THE THERMAL/HYDRAULIC ANALYSIS OF LMFBR CHECK VALVES. Cho, S.M.; Kane, R.S. (Foster Wheeler Energy Corp., Livingston, NJ (USA); Manhattan Coll., Riverdale, NY (USA). Dept. of Mechanical Engineering). 1979. Contract EY-76-C-15-2395. 25p. Dep. NTIS, PC A02/MF A01.

From ASME winter annual meeting; New York, NY, USA (2 Dec 1979).

A thermal/hydraulic analysis of the check valves in liquid sodium service for LMFBR plants is required to provide temperature data for thermal stress analysis of the valves for specified transient conditions. Because of the complex three-dimensional flow pattern within the valve, the heat transfer analysis techniques for less complicated shapes could not be used. This paper discusses the thermal analysis techniques used to assure that the valve stress analysis is conservative. These techniques include a method for evaluating the recirculating flow patterns and for selecting appropriately conservative heat transfer correlations in various regions of the valve.

- 847 (CONF-791205--18) DESIGN AND OPERATING EXPERIENCE OF EBR-II INTERMEDIATE HEAT EXCHANGER. Buschman, H.W.; Koenig, J.F.; Stone, C.C. (Argonne National Lab., Idaho Falls, ID (USA); Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 23p. Dep. NTIS, PC A02/MF A01.

From ASME winter annual meeting; New York, NY, USA (2 Dec 1979).

Experimental Breeder Reactor II (EBR-II) has operated for over 15 years at the Idaho National Engineering Laboratory near Idaho Falls. EBR-II has served the nation in providing information on fuels, materials, and components under conditions approaching those expected for commercial power plants. In addition, EBR-II is a power plant generating electricity with an availability of about 70%. A key component, the Intermediate Heat Exchanger (IHX), of any Liquid Metal Fast Breeder Reactor (LMFBR) must perform with a high degree of reliability for successful commercialization. The design and operating experience gained from EBR-II demonstrates that the IHX can be built and operated with confidence that its performance and reliability will be satisfactory.

- 848 (CONF-791205--23) HEAT-TRANSFER IN A PARTIALLY-BLOCKED SODIUM-COOLED ROD BUNDLE. Han, J.T. (Oak Ridge National Lab., TN

(USA)). 1979. Contract W-7405-ENG-26. 6p. NTIS, PC A02/MF A01.

From ASME winter annual meeting; New York, NY, USA (2 Dec 1979).

Heat transfer coefficients were experimentally determined for 31-rod sodium-cooled bundle with a 6-subchannel central blockage. The Nusselt number is presented as a function of the Peclet number for both the free flow region undisturbed by the blockage and the wake region immediately downstream of the blockage. Results are compared with the existing correlations for liquid metals. The heat transfer coefficient was generally higher in the unblocked free flow region than in the wake region. A leak at the blockage improved the heat transfer coefficient in the wake region.

849 (CONF-800226--1) BUOYANCY INDUCED FLOW AND HEAT REDISTRIBUTION DURING LMFBR CORE DECAY HEAT REMOVAL. Coffield R.D. Jr.; Tang, Y.S.; Killimayer, J.S.; Markley, R.A. (Westinghouse Electric Corp., Madison, PA (USA)). 1980. Contract AC15-76CL02395. 14p. Dep. NTIS, PC A02/MF A01.

From Specialists meeting on decay heat removal and natural circulation in FBR's; Upton, NY, USA (28 Feb 1980).

Natural convection cooling of the core exhibits one of the few core design transients where low power/high temperature conditions exist. Core inter-assembly and intra-assembly flow/heat redistribution: (a) become significant with regard to accurately predicting temperatures; and (b) decrease the maximum hot element temperatures. A technique is described herein for linking dynamic core-wide transient effects as predicted by COBRA-WC to the localized hot element temperatures calculated with FORE-2M. These latter temperatures are used in evaluations with regard to core design adequacy.

850 (CONF-800226--6) RESPONSE OF EBR-II TO A COMPLETE LOSS OF PRIMARY FORCED FLOW DURING POWER OPERATION. Singer, R.M.; Gillette, J.L.; Mohr, D.; Tokar, J.V.; Sullivan, J.E.; Dean, E.M. (Argonne National Lab., IL (USA)). 1980. Contract W-31-109-ENG-38. 14p. Dep. NTIS, PC A02/MF A01.

From Specialists meeting on decay heat removal and natural circulation in FBR's; Upton, NY, USA (28 Feb 1980).

Detailed measurements of the thermal, hydraulic, and neutronic response of EBR-II to a complete loss of primary forced flow followed by a PPS-activated scram are presented. The experimental results clearly indicate a smooth transition to natural convective flow with a quite modest incore temperature transient. The accompanying calculations using the NATDEMO code agree quite well with the measured temperatures and flow rates throughout the primary system. The only region of the plant where a significant discrepancy between the measurements and calculations occurred was in the IHX. The reasons for this result could not be definitively determined, but it is speculated that the one-dimensional assumptions used in the modeling may not be valid in the IHX during buoyancy driven flows.

851 (CONF-800334--7) DEVELOPMENTS IN EFFLUENT CONTROL FOR BREEDER REPROCESSING FACILITIES. Yarbrow, G.O.; Sears, M.B. (Oak Ridge National Lab., TN (USA)). 1980. Contract W-7405-ENG-26. 8p. Dep. NTIS, PC A02/MF A01.

From 2. DOE environmental control symposium; Reston, VA, USA (17 Mar 1980).

10 CFR 20 and 40 CFR 190 impose tight restrictions on the releases of radioactive materials from the fuel cycle. Retention factors needed in the LMFBR and LWR fuel cycles to meet the 40 CFR 190 release limits of ^{85}Kr , ^{129}I , and ^{239}Pu are listed. Technologies for the control of I, ^{85}Kr , and tritium are discussed. Advanced effluent control systems will be demonstrated in the Hot Experimental Facility. (DLC)

852 (CONF-800401--5) EMISSIVITY OF SODIUM WETTED AND OXIDIZED TYPE 304 STAINLESS STEEL. Haines, N.L.; Craig, R.E.; Forsyth, D.R.; Novendstern, E.H. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1980. Contract AC15-76CL02395. 7p. Dep. NTIS, PC A02/MF A01.

From 2. international conference of liquid metal technology for energy systems; Richland, WA, USA (20 Apr 1980).

The emissivity of sodium wetted and oxidized Type 304 stainless steel was determined to provide data for calculating the heat flow through Liquid Metal Fast Breeder Reactor (LMFBR) reflector plates, located above the sodium pool, to the reactor closure head. An emissivity experiment using a Type 304 stainless steel specimen was performed in an inerted glovebox. Relatively high oxygen concentrations of 10,000 and 50 vppm were used in the argon/oxygen mixtures to reduce reaction time. Following wetting and oxidation, the specimen was heated to a maximum temperature of 450°C and the emissivity of the oxidized coating was calculated. Results indicate that the emissivity of the coating ranged from 0.55 to 0.92.

853 (CONF-800434--2) ORNL EXPERIMENTS TO CHARACTERIZE FUEL RELEASE FROM THE REACTOR PRIMARY CONTAINMENT IN SEVERE LMFBR ACCIDENTS. Wright, A.L.; Kress, T.S.; Smith, A.M. (Oak Ridge National Lab., TN (USA)). 1980. Contract W-7405-ENG-26. 16p. Dep. NTIS, PC A02/MF A01.

From CSNI specialist meeting on nuclear aerosols in reactor safety; Gatlinburg, TN, USA (15 Apr 1980).

This paper presents results from aerosol source term experiments performed in the ORNL Aerosol Release and Transport (ART) Program sponsored by the US NRC. The tests described were performed to provide information on fuel release from an LMFBR primary containment as a result of a hypothetical core-disruptive accident (HCDA). The release path investigated in these tests assumes that a fuel/sodium bubble is formed after disassembly that transports fuel and fission products through the sodium coolant and cover gas to be released into the reactor secondary containment. Due to the excellent heat transfer characteristics of the sodium, there is potential for large attenuation of the maximum release.

854 (CONF-800434--3) BEHAVIOR OF SODIUM OXIDE, URANIUM OXIDE AND MIXED SODIUM OXIDE-URANIUM OXIDE AEROSOLS IN A LARGE VESSEL. Adams, R.E.; Han, J.T.; Kress, T.S.; Silberberg, M. (Oak Ridge National Lab., TN (USA)). 1980. Contract W-7405-ENG-26. 20p. Dep. NTIS, PC A02/MF A01.

From CSNI specialist meeting on nuclear aerosols in reactor safety; Gatlinburg, TN, USA (15 Apr 1980).

Results are reported for experiments on the behavior of sodium oxide aerosols, uranium oxide aerosols, and various mixtures of these in the Nuclear Safety Pilot Plant (NSPP) vessel at Oak Ridge National Laboratory. The NSPP

program, sponsored by the Nuclear Regulatory Commission, is to provide experimental qualification of the HAARM-3 aerosol behavioral code being independently developed at Battelle-Columbus Laboratories. The results of fourteen single-component and six mixed-component aerosol experiments are presented along with selected comparisons with code predictions.

- 855 (CONF-800607--5) INFLUENCE OF CONTROL ROD WORTH INTERACTIONS ON LMFBR CONTROL SYSTEMS DESIGN. Lake, J.A.; Rittenberger, R.V.; Rathbun, R.W. (Westinghouse Electric Corp., Pittsburgh, PA (USA). Advanced Reactors Div.). 1980. Contract AC15-76CL02395. 5p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

These design criteria are interpreted to define the reactivity worth requirements for the primary and secondary control systems in terms of the minimum control systems capability under faulted conditions which will assure that the reactor can be safely shut down. The faulted conditions are postulated to occur upon the simultaneous failure of one of the redundant safety control systems to scram, a stuck rod in the scrambling system, and a reactivity insertion resulting from the uncontrolled withdrawal of the highest worth control rod inserted in the reactor. The resulting positive reactivity insertion from the rod runout envelopes other postulated operational faults and is imposed on the shutdown requirements of both the primary and secondary control systems. In order to determine the minimum shutdown capability, an evaluation is made of the worst combination of control rod runout (reactivity fault) and stuck rod worth.

- 856 (CONF-800607--28) IRRADIATED TYPE 304 SS BEND TEST DUCTILITY. Garkisch, H.D. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1980. Contract AC15-76CL02395. 6p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

A series of tests were performed to determine the available bending deformation capability and ductility of preirradiated Type 304 stainless steel (SS) for core applications. This material was tested under conditions at which it exhibits less ductility than Type 316 SS or other stainless steels currently under considerations for core applications. Tensile testing of Type 304 SS with specimens from EBR-II ducts which were irradiated to fluences between 5 to 10 x 10²² n/cm² (E > 0.1 MeV) revealed a significant reduction of uniform elongation (UE) and total elongation (TE), but smaller decreases in reduction of area (RA). These specimens fail with significant but very localized necking. The ductility of this material decreases with decreasing strain rate if the test temperature is above the irradiation temperature. A large reduction in ductility is caused by a change from a transgranular to an intergranular fracture mode.

- 857 (CONF-800607--29) BLANKET DESIGN CONSIDERATIONS FOR FAST REACTOR OXIDE FUEL SYSTEMS BASED ON CRBRP EXPERIENCE. Garkisch, H.D. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1980. Contract AC15-76CL02395. 6p. Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

The blanket assemblies are of prime importance in achieving the performance goals of fast reactor fuel systems to conserve uranium ore and to minimize fuel cycle costs. The purpose of past blanket design efforts in the US was to meet these goals with designs of optimum fuel and thermal performance. Only the Clinch River Breeder Reactor Plant (CRBRP) blanket design has evolved beyond the conceptual stage. The CRBRP blanket assembly design is based on conservatively predicted environments and criteria which are being confirmed by a design oriented development program. The results of the ongoing program obtained to date have not indicated any need for assembly design changes. However, since the irradiation testing of blanket rods is not completed, the ultimate performance capability of the blanket is presently not known. Therefore, conservative blanket assembly design criteria and predicted environments must be utilized for performance predictions at the present time.

- 858 (CONF-800607--33) FINAL DESIGN ANALYSES OF CRBRP FUEL ROD. Bishop, B.A.; Vijuk, R.M. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1980. Contract AC15-76CL02395; EY-76-C-15-0003. 6p. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

The key results are presented of the CRBRP fuel rod analyses performed in support of the CRBRP fuel assembly final design review in December, 1977. For the CRBRP fuel rod design environment, it is shown that the life limiting criteria on ductility limited strain (DLS) and cumulative damage function (CDF) are satisfied. The conservatism in these analyses is demonstrated by comparison to the upset enthalpy based failure potential criteria derived from cladding breach experience for TREAT simulated untempered rapid reactivity insertion transients.

- 859 (CONF-7706129--), pp 8p, Paper 1) CLINCH RIVER BREEDER REACTOR AND ENERGY INDEPENDENCE. Anderson, J.C. (Oak Ridge Associated Universities, TN). 1977.

From AIPE annual convention proceedings; Atlanta, GA, USA (20 Jun 1977).

Nuclear energy with special emphasis on the Clinch River Breeder Reactor (CRBRP) program is discussed. The CRBRP is an important demonstration project in that it will produce electricity and make fuel. For every twelve atoms of fuel utilized to produce heat/electricity, twelve new atoms of fuel will be manufactured. This demonstration will point toward a method of "manufacturing" fuel at a time when naturally-occurring fossil fuels and uranium-235 are being depleted by an ever increasing number of countries each year. Both fossil and uranium-235 fuels are limited. As these finite fuel supplies vanish, it is important that additional energy sources be developed.

- 860 (CONF-7706130--), pp 12p, Session 2, Paper 1) FUTURE OF THE BREEDER IN THE U.S. Iacobellis, S.F. (Atomics International Div., Canoga Park, CA). 1977.

From Conference on U.S. options for long term energy supply; Denver, CO, USA (19 Jun 1977).

The current political and social forces affecting the U.S. LMFBR program are discussed. The need for timely LMFBR development is emphasized.

- 861 (CONF-7706130--), pp 17p, Session 2, Paper 2) LMFBRs: WHERE DO WE GO FROM HERE. Beckjord, E.S. (Energy Research and Development Administration, Washington, DC). 1977.

From Conference on U.S. options for long term energy supply; Denver, CO, USA (19 Jun 1977).

The mission and current status of the U.S. LMFBR program are discussed. Changes in the program made by the Carter administration are described, and the direction and pace of future activities are discussed.

- 862 (CONF-7706130--), pp 23p, Session 1, Paper 3) ALTERNATE PERSPECTIVE ON LONG-RANGE ENERGY OPTIONS. Culler, F.L. (Oak Ridge National Lab., TN). 1977.

From Conference on U.S. options for long term energy supply; Denver, CO, USA (19 Jun 1977).

The author emphasizes that, in his opinion and that of the industrial segment of the economic establishment, nuclear energy will be needed to replace in part the rapidly diminishing supplies of oil and gas. To maintain the nuclear option, breeders are necessary, he continues. With uranium of recoverable grade in limited supply Mr. Culler says that prudent planning should include the demonstration of the breeder option, which must also provide efficient and safe reprocessing, fuel recycle, and radioactive waste disposal. (MCW)

- 863 (CONF-7711121--), PROCEEDINGS OF THE INDO-GERMAN SEMINAR ON RADIATION DAMAGE. (Reactor Research Centre, Kalpakkam (India)). 1977. 521p. Dep. NTIS (US Sales Only), PC A22/MF A01.

From Indo-German seminar on radiation damage; Kalpakkam, India (14 Nov 1977).

Information is presented concerning fundamental aspects of radiation damage; voids, creep, and other mechanical properties; applications to reactor materials; and irradiation studies and facilities. (FS)

- 864 (CONF-7810133--5) ENVIRONMENTALLY IMPORTANT RADIONUCLIDES IN NONPROLIFERATIVE FUEL CYCLES. Kaye, S.V.; Till, J.E. (Oak Ridge National Lab., TN (USA)). 1978. Contract W-7405-ENG-26. 22p. Dep. NTIS, PC A02/MF A01.

From 12. annual symposium of the German Health Physics Society--radioactivity and the environment; Norderney, F.R. Germany (2 Oct 1978).

Our analyses indicate that more in-depth research should be done on ^3H , ^{14}C , ^{99}Tc , and ^{232}U , especially because of their presence in nonproliferative fuel cycles. For increased ^3H production by fast reactors, we can only speculate that such research could show that environmental releases might be significantly greater than for LWRs. Carbon-14 will likely not be a problem if a suitable decontamination factor can be agreed upon for reprocessing facilities and if a satisfactory regulatory limit can be established for global populations. Additional experimental research is urgently needed to determine the uptake of low levels of ^{99}Tc by plants. These data are essential before an accurate assessment of ^{99}Tc releases can be made. Finally, we recommend that investigators take a closer look at the potential problems associated with ^{232}U and daughters. This radionuclide could contribute a significant portion of the dose in both environmental and occupational exposures from

the nonproliferative fuels.

- 865 (CONF-7910112--2) CURRENT STATUS OF FAST REACTOR PHYSICS. Hummel, H.H. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 46p. Dep. NTIS, PC A03/MF A01.

From NEACRP meeting; Paris, France (1 Oct 1979).

The subject of calculation of reactivity coefficients for fast reactors is developed, starting with a discussion of the status of relevant nuclear data and proceeding to the subjects of group cross section generation and of methods of obtaining reactivity coefficients from group cross sections. Reactivity coefficients measured in critical experiments are compared with calculated values. Dependence of reactivity coefficients on reactor design is discussed. Finally, results of the recent international comparison of calculated reactivity coefficients are presented.

- 866 (COO--2172-29) MECHANICAL PROPERTIES OF CRYSTALLINE SOLIDS. PROGRESS REPORT, DECEMBER 1, 1978-NOVEMBER 30, 1979. Li, C.; Hart, E.W. (Cornell Univ., Ithaca, NY (USA). Dept. of Materials Science and Engineering). 1 Sep 1979. Contract EY-76-S-02-2172. 60p. Dep. NTIS, PC A04/MF A01.

The work performed during the past fiscal year included: the development and preliminary experimental verification of a second generation deformation model based on state variables, which more correctly represents transient deformation; the establishment of experimental capabilities for investigating grain boundary anelasticity and the development of unified phenomenological models for grain boundary anelasticity and grain boundary sliding; the continued investigation of grain boundary cavitation processes in the Zircalloys; and the theoretical modeling of stress relaxation in thin films.

- 867 (COO--2245-17TR(Rev.1)) INPUT PARAMETERS TO CODES WHICH ANALYZE LMFBR WIRE WRAPPED BUNDLES. REVISION 1. Wang, S.F.; Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA)). May 1979. Contract EY-76-S-02-2245. 57p. Dep. NTIS, PC A04/MF A01.

This report provides a current summary of recommended values of key input parameters required for ENERGY code analysis of LMFBR wire wrapped bundles. This data is based on the interpretation of experimental results from the MIT and other available laboratory programs.

- 868 (COO--2245-27TR(Rev.1)) FABRICATION DETAILS FOR WIRE WRAPPED FUEL ASSEMBLY COMPONENTS. Bosy, B.J. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). Sep 1978. Contract EY-76-S-02-2245. 64p. Dep. NTIS, PC A04/MF A01.

Revised by H. Khan.

Extensive hydraulic testing of simulated LMFBR blanket and fuel assemblies is being carried out under this MIT program. The fabrication of these test assemblies has involved development of manufacturing procedures involving the wire wrapped pins and the flow housing. The procedures are described in detail in the report.

- 869 (COO--2245-35TR) ANALYSIS OF THE FEASIBILITY OF IMPLEMENTING AN IMPLICIT TEMPORAL DIFFERENCING SCHEME IN THE SUPERENERGY CODE. Masterson, R.; Todreas, N. (Massachusetts Inst. of Tech., Cambridge (USA)).

Dept. of Nuclear Engineering). Feb 1977.
Contract EY-76-S-02-2245. 34p. Dep. NTIS,
PC A03/MF A01.

In order to assess the optimum approach for extending the SUPERENERGY code to transient LMFBR analysis, it is necessary to consider the advantages and disadvantages of the various temporal differencing schemes that can be applied to the energy equation used by the code. These differencing schemes can be fully implicit or fully explicit, although differencing schemes consisting of a combination of temporally implicit and explicit parts may also be used. It is well recognized that temporally implicit differencing schemes tend to be more stable, but that this stability can be offset in practice by factors such as greater computer storage requirements and considerably longer running times.

- 870 (COO--2245-36TR) COLLECTION AND EVALUATION OF SALT MIXING DATA WITH THE REAL TIME DATA ACQUISITION SYSTEM. Glazer, S.; Chiu, C.; Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA)). Dept. of Nuclear Engineering). Sep 1977. Contract EY-76-S-02-2245. 331p. Dep. NTIS, MF A01.

Portions of document are illegible.

A minicomputer based real time data acquisition system was designed and built to facilitate data collection during salt mixing tests in mock ups of LMFBR rod bundles. The system represents an expansion of data collection capabilities over previous equipment. It performs steady state and transient monitoring and recording of up to 512 individual electrical resistance probes. Extensive real time software was written to govern all phases of the data collection procedure, including probe definition, probe calibration, salt mixing test data acquisition and storage, and data editing. Offline software was also written to permit data examination and reduction to dimensionless salt concentration maps. Finally, the computer program SUPERENERGY was modified to permit rapid extraction of parameters from dimensionless salt concentration maps. The document describes the computer system, and includes circuit diagrams of all custom built components. It also includes descriptions and listings of all software written, as well as extensive user instructions.

- 871 (COO--2245-37TR) APPROXIMATE METHOD FOR PREDICTING TEMPERATURE DISTRIBUTION IN WIRE WRAPPED FUEL ASSEMBLIES OF A LMFBR. Mikic, B.; Khan, E.U.; Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA)). Dept. of Nuclear Engineering). Apr 1977. Contract EY-76-S-02-2245. 31p. Dep. NTIS, PC A03/MF A01.

In Liquid Metal Fast Breeder Reactors (LMFBR) the fuel and blanket assemblies are subject to radial gradients in the rate of heat generations and have a wide range of flow conditions. An approximate method is presented for predicting temperature distributions in wire wrapped fuel assemblies based on a porous body model which was previously formulated by Khan. Numerical techniques were used for the solution of the equation set. However it was indicated that an analytical solution is possible for the case where the bundle is operating in forced convection only (negligible free convection). An analytical method is presented which, when combined with a very simple graphical construction, provides an engineering estimate of the temperature distribution in wire wrapped fuel rod assemblies.

- 872 (COO--2245-40TR) WARD BLANKET ASSEMBLY PRE-TEST PREDICTIONS BY SUPERENERGY. Chiu, C.; Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA)). Dept. of Nuclear Engineering). May 1977. Contract EY-76-S-02-2245. 67p. Dep. NTIS, PC A04/MF A01.

A heated LMFBR blanket assembly is under test at WARD to determine its thermal-hydraulic characteristics. Predictions of part of the WARD test results using the SUPERENERGY code with the aid of existing flow split and salt mixing data are presented. A significant part of this effort is the detailed consideration of the uncertainties introduced in temperature field predictions due to as-built bundle tolerances.

- 873 (COO--2245-41TR) FLOW SPLIT MEASUREMENTS IN LMFBR BLANKET ASSEMBLIES. Chiu, C.; Todreas, N.E.; Rohsenow, W.M. (Massachusetts Inst. of Tech., Cambridge (USA)). Apr 1978. Contract EY-76-S-02-2245. 136p. Dep. NTIS (US Sales Only), PC A07/MF A01.

Individual subchannel flow rates are isokinetically measured for all the edge and interior subchannels (at a series of Reynolds numbers ranging from laminar to turbulent) in two different LMFBR blanket assemblies. The specific goals of this task are to obtain the average subchannel flow rates for both interior and edge subchannels in the bundle ($P/D = 1.067$, $H/D = 8.0$) which is geometrically similar to the blanket assemblies ($P/D = 1.077$, $H/D = 7.7$) used in the Clinch River Breeder Reactor and to obtain the information of the lead length effect on the average subchannel flow rates through the subchannel flow rate measurement in the bundle which has $P/D = 1.067$ and $H/D = 4.0$.

- 874 (COO--2245-42TR) PRESSURE DROP MEASUREMENTS IN LMFBR WIRE WRAPPED BLANKET ASSEMBLIES. Chiu, C.; Hawley, J.; Rohsenow, W.M.; Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA)). Jul 1977. Contract EY-76-S-02-2245. 77p. Dep. NTIS, PC A05/MF A01.

In this experiment, measurements of subchannel static pressure for an interior and edge subchannel were taken at two elevations in two wire-wrapped 61-pin bundles. One of the bundles has geometric characteristics of $P/D = 1.067$ and $H/D = 8.0$ (4 inch lead length and 0.501 inch rod diameter) and the other bundle has geometric characteristics of $P/D = 1.067$ and $H/D = 4.0$ (2 inch lead length and 0.501 inch rod diameter). The bundle average friction factors as well as the local subchannel friction factors for both interior and edge subchannels were determined from the experimental static pressure data. The average subchannel flow rates for both edge and interior subchannels were determined in a separate experiment. Results show that two correlations suggested by Rehme and Novendstern for the bundle average friction factor cannot predict the data within the range of experimental error. The bundle average friction factors for both bundles under test were underestimated by Rehme's correlation and overestimated by Novendstern's correlation. The results of the local subchannel friction factors indicate the effect of the wire lead length is more pronounced in the interior subchannel friction factor than in the edge subchannel friction factor. As the wire wrap lead length decreases, both interior and edge subchannel friction factors increase.

875 (COO--2245-43TR) MIXING EXPERIMENTS IN LMFBR WIRE WRAPPED BLANKET ASSEMBLIES. Chib, C.; Rohsenow, W.M.; Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA)). Apr 1978. Contract EY-76-S-02-2245. 173p. Dep. NTIS (US Sales Only), PC A08/MF A01.

The saline water injection technique is used to obtain the salt concentration maps at the exits of two LMFBR 61-pin blanket assemblies. The intersubchannel mixing parameter ϵ_{sub}/l and the swirl flow parameter C_{sub}/l which are inputs to the SUPERENERGY (or ENERGY) computer code are reduced from the exit salt concentration maps. Results show that both parameters are independent of the bundle Reynolds number under the high turbulent flow conditions and decrease as the Reynolds number decreases into the laminar flow regime. Both parameters for the four inch lead bundles are less than those for the two inch lead bundle by a factor of 1.8.

876 (COO--2245-44TR) COOLANT MIXING IN THE LMFBR OUTLET PLENUM. Chen, Y.B.; Golay, M.W. (Massachusetts Inst. of Tech., Cambridge (USA)). Dept. of Nuclear Engineering. Jun 1977. Contract EY-76-S-02-2245. 224p. Dep. NTIS, PC A10/MF A01.

Small scale experiments involving water flows are used to provide mean flow and turbulence field data for LMFBR outlet plenum flows. Measurements are performed at Reynolds Number (Re) values of 33000 and 70000 in a 1/15-scale FFTF geometry and at Re = 35000 in a 3/80-scale CRBR geometry. The experimental behavior is predicted using two different turbulence model computer programs, TEACH-T and VARR-II. It is found that the qualitative nature of the flow field within the plenum depends strongly upon the distribution of the mean inlet velocity field, upon the degree of inlet turbulence, and upon the turbulence momentum exchange model used in the calculations. It is found in the FFTF geometry that the TEACH-T predictions are better than that of VARR-II, and in the CRBR geometry neither code provides a good prediction of the observed behavior. From the sensitivity analysis, it is found that the production and dissipation of turbulence are the dominant terms in the transport equations for turbulent kinetic energy and turbulent energy dissipation rate, and the diffusion terms are relatively small. From the same study a new set of empirical constants for the turbulence model is evolved for the prediction of plenum flows.

877 (COO--2245-45TR) TURBULENT INTERCHANGE IN TRIANGULAR ARRAY BARE ROD BUNDLES. Kelly, J.M.; Todreas, N. (Massachusetts Inst. of Tech., Cambridge (USA)). Dept. of Nuclear Engineering. Jul 1977. Contract EY-76-S-02-2245. 227p. Dep. NTIS, PC A11/MF A01.

Bulk mixing coefficients were measured for single plane water flow in a simulated rod bundle with a pitch to diameter ratio of 1.10. A tracer technique employing Rhodamine B as the tracer and measuring fluorescence was used. Isokinetic sampling was achieved by using a pressure balance method. The results were corrected for both entrance effects and diversion crossflows. The results showed a change in Reynolds number behavior as the laminar sublayer began to "choke" the turbulent mixing. This, and a review of other mixing experiments, suggested that secondary flows do not compensate for laminarization and that turbulent mixing decreases as the pitch to diameter ratio decreases for values of P/D less than 1.05 in a manner similar to that predicted by Ramm et al. Concentration profiles were

measured through the clearance gap and the values of the gradient were used to calculate the gap averaged circumferential eddy diffusivity for mass. A discussion of the eddy diffusivity concept and its applicability to turbulent mixing is presented.

878 (COO--2245-46TR) ASSESSMENT OF THE NEED TO INCORPORATE THE VARIABLE SWIRL MODEL IN THE ENERGY CODE. Basehore, K.L.; Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA)). Dept. of Nuclear Engineering. Jul 1977. Contract EY-76-S-02-2245. 74p. Dep. NTIS, MF A01.

Portions of document are illegible.

At present the ENERGY model contains a correlated swirl flow that is circumferentially and axially constant. This report evaluates the need to include a variable swirl flow forcing function in this model. A suitable function for testing the variable swirl flow effects is obtained by reducing the existing world transverse flow experimental data, relying heavily on the MIT laser doppler anemometer data. A comparison of the existing constant swirl model with the variable swirl model using the resulting forcing function is made with the help of a few sample problems.

879 (COO--2245-47TR) ANALYSIS OF THE THERMAL-HYDRAULIC BEHAVIOR IN THE CRBR SECONDARY CONTROL ASSEMBLY, INCLUDING INTERASSEMBLY HEAT TRANSFER EFFECTS. Basehore, K.L.; Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA)). Dept. of Nuclear Engineering. Jul 1977. Contract EY-76-S-02-2245. 173p. Dep. NTIS, PC A08/MF A01.

Portions of document are illegible.

A first effort is presented at modeling the thermal hydraulic behavior of the CRBR secondary control assembly (SCA). For modeling purposes, the SCA is broken into two sections, the counterflow section and the poison control section, which contains the control bundle. The coolant temperature distribution in the counterflow section is evaluated by using a semi-analytical solution technique. The distribution in the poison control section is found by using forward differenced numerical techniques. The wire wrap sweeping in the control bundle is represented by an empirical enhanced eddy diffusion of heat as in the ENERGY model.

880 (COO--2245-48TR) HYDRODYNAMIC BEHAVIOR OF A BARE ROD BUNDLE. Bartzis, J.G.; Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA)). Dept. of Nuclear Engineering. Jun 1977. Contract EY-76-S-02-2245. 370p. Dep. NTIS, PC A16/MF A01.

Thesis. Submitted by J.G. Bartzis.

Portions of document are illegible.

The temperature distribution within the rod bundle of a nuclear reactor is of major importance in nuclear reactor design. However temperature information presupposes knowledge of the hydrodynamic behavior of the coolant which is the most difficult part of the problem due to complexity of the turbulence phenomena. In the present work a 2-equation turbulence model--a strong candidate for analyzing actual three dimensional turbulent flows--has been used to predict fully developed flow of infinite bare rod bundle of various aspect ratios (P/D). The model has been modified to take into account anisotropic effects of eddy viscosity. Secondary flow calculations have been also performed although the model seems to be too rough to predict the secondary flow correctly. Heat transfer calculations have been

performed to confirm the importance of anisotropic viscosity in temperature predictions. All numerical calculations for flow and heat have been performed by two computer codes based on the TEACH code. Experimental measurements of the distribution of axial velocity, turbulent axial velocity, turbulent kinetic energy and radial Reynolds stresses were performed in the developing and fully developed regions. A 2-channel Laser Doppler Anemometer working on the Reference mode with forward scattering was used to perform the measurements in a simulated interior subchannel of a triangular rod array with $P/D = 1.124$. Comparisons between the analytical results and the results of this experiment as well as other experimental data in rod bundle array available in literature are presented. The predictions are in good agreement with the results for the high Reynolds numbers.

- 881 (COO--2245-49TR) MEASUREMENT AND ANALYSIS OF FLOW WALL SHEAR STRESS IN AN INTERIOR SUBCHANNEL OF TRIANGULAR ARRAY RODS. Fakori-Monazah, M.R.; Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). Aug 1977. Contract EY-76-S-02-2245. 206p. Dep. NTIS, PC A10/MF A01.

Thesis.

A simulated model of triangular array rods with pitch to diameter ratio of 1.10 (as a test section) and air as the fluid flow was used to study the LMFBR hydraulic parameters. The wall shear stress distribution around the rod periphery, friction factors, static pressure distributions and turbulence intensity corresponding to various Reynolds numbers ranging from 4140 to 36170 in the central subchannel were measured. Various approaches for measurement of wall shear stress were compared. The measurement was performed using the Preston tube technique with the probe outside diameter equal to 0.014 in.

- 882 (COO--2245-50) COOLANT MIXING IN LMFBR ROD BUNDLES AND OUTLET PLENUM MIXING TRANSIENTS. PROGRESS REPORT, MARCH 1, 1977--MAY 31, 1977. Todreas, N.E.; Golay, M.W.; Wolf, L. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 1977. Contract EY-76-S-02-2245. 100p. Dep. NTIS, PC A05/MF A01.

Progress is summarized in the following tasks: (1) bundle flow studies (wrapped and bare rods); (2) subchannel flow studies (bare rods); (3) LMFBR outlet plenum flow mixing; and (4) theoretical determination of local temperature fields in LMFBR fuel rod bundles. (DG)

- 883 (COO--2245-51TR) FLUID MIXING STUDIES IN A HEXAGONAL 61-PIN, WIRE-WRAPPED ROD BUNDLE. Hanson, A.S.; Todreas, N. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). Aug 1977. Contract EY-76-S-02-2245. 568p. Dep. NTIS, PC A14/MF A01.

Two wire-wrapped rod bundles with different leads (6 in. and 12 in.) were constructed with geometric parameters similar to proposed LMFBR fuel assemblies. Rod diameter was 0.25 in. and pitch-to-diameter ratio was 1.26. These two bundles were tested in a flow loop which was designed and built for mixing experiments. Fluid mixing was studied by means of salt tracer dispersion. Salt was injected at various radial and axial locations in the bundle via injection rods, and then the dispersed distribution was measured at the bundle exit by

means of 126 specially designed electrical conductivity probes inserted into the bundle subchannels. The data collected showed a strong swirl flow around the bundle circumference and periodic variation with axial injection location. Data from turbulent runs was generally good with mass balances averaging 90% and having a spread of $\pm 25\%$. The laminar data collected was generally poor because of a "striping" phenomena and injection instabilities. Data were compared with calculations using the ENERGY computer code. The comparison between ENERGY calculations and the data was not good for laminar flow and was only fair in the turbulent cases. It was found that turbulent data could be best characterized by the ENERGY parameters $C_1 = 0.19$ and $\epsilon_{1,*} = 0.025$ when the lead was 6 inches; for a 12-inch lead the parameters were $C_1 = 0.16$ and $\epsilon_{1,*} = 0.012$. Pressure drop data was also taken from the two bundles and it too showed a periodic variation with axial location. Friction factors derived from the data were generally higher than predicted by available correlations. These data suggested that traditional flow split calculations could be in error and that the laminar-turbulent transition occurs over a broad Reynolds number range in wire-wrapped rod bundles.

- 884 (COO--2245-53) COOLANT MIXING IN LMFBR ROD BUNDLES AND OUTLET PLENUM MIXING TRANSIENTS. PROGRESS REPORT, JUNE 1, 1977--AUGUST 31, 1977. Todreas, N.E.; Golay, M.W.; Wolf, L. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 1977. Contract EY-76-S-02-2245. 32p. Dep. NTIS, MF A01.

Portions of document are illegible.

Information is presented concerning coolant mixing in bundle geometries of wrapped and bare rods, mixing in subchannel geometry of bare rods, and LMFBR outlet plenum flow mixing.

- 885 (COO--2245-54TR) MIXING EXPERIMENTS IN AN ALTERNATING WIRE WRAPPED ASSEMBLY. Chiu, C.; Todreas, N.E.; Rohsenow, W.M. (Massachusetts Inst. of Tech., Cambridge (USA)). Dec 1977. Contract EY-76-S-02-2245. 201p. Dep. NTIS, PC A10/MF A01.

The salt injection experiment was performed in an alternating wire-wrapped, triangular-array bundle to study the coolant mixing behavior pertaining to alternating wrapped wires. Results show that the coolant mixing is much enhanced in the subchannels for this type of bundle compared to the in-phase wire-wrapped bundle currently used in the LMFBR design. A strong interaction between the coolants in the edge and the interior subchannels at four assembly faces (where the alternating wires give 180° out-of-phase configuration) has also been observed. After calibrating the input parameters for the sweeping flow of the COBRA-IIIC/MIT code against this experimental data, the code is employed to predict the coolant temperatures for a skew-powered alternating wire-wrapped blanket assembly. Comparing the code results with the SUPERENERGY code predicted coolant temperature data for an in-phase wrapped assembly under identical operating conditions, it is observed that the hot spot temperature for the alternating wire-wrapped assembly is less than that for the in-phase wire-wrapped bundle by 5 percent (normalized to the hot spot axial temperature rise).

- 886 (COO--2245-55TR) TURBULENT SWEEPING FLOW MIXING MODEL FOR WIRE WRAPPED LMFBR ASSEMBLIES. Chiu, C.; Rohsenow, W.M.;

Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA)). Apr 1978. Contract EY-76-S-02-2245. 69p. Dep. NTIS (US Sales Only), PC A04/MF A01.

A physical model is proposed to derive the sweeping flow for LMFBR triangular array wire-wrapped assemblies under the turbulent flow condition. Two correlations are suggested for the sweeping flow through two different types of gaps between subchannels, the gap between the interior subchannels and the gap between the wall subchannels. These two sweeping flow correlations are evolved by calibrating the constants in the proposed model against the available experimental data. Agreement between the correlations and all the experimental data of ± 35 percent is obtained over the assembly design range of 1.315 greater than P/D greater than 1.067 and 52 greater than H/D greater than 4 . Based on these correlations, flow sweeping input parameters for two popular computer codes, i.e., COBRA and SUPERENERGY, are recommended.

TRANSFER ANALYSIS OF COUPLED COOLANT CELLS IN FINITE LMFBR BUNDLES. Wong, C.N.; Wolf, L. (Massachusetts Inst. of Tech., Cambridge (USA)). Dept. of Nuclear Engineering). Feb 1978. Contract EY-76-S-02-2245. 327p. Dep. NTIS, PC A15/MF A01.

A three-dimensional single region slug flow heat transfer analysis for finite LMFBR rod bundles using a classical analytical solution method has been performed. According to the isolated single cell analysis, the results show that the peripheral clad temperature variation as well as the thermal entrance length are strongly dependent upon the degree of irregularity displayed by various coolant geometries. Since under the present LMFBR conditions, fully-developed temperature fields may hardly be established in such characteristic rod bundle regions, a 3-D heat transfer analysis seems to be mandatory. This implies that the results of fully developed heat transfer analyses are by far too conservative.

887 (COO--2245-55TR(Rev.1)) TURBULENT SWEEPING FLOW MIXING MODEL FOR WIRE WRAPPED LMFBR ASSEMBLIES. Chiu, C.; Rohsenow, W.M.; Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA)). Dept. of Nuclear Engineering). Oct 1978. Contract EY-76-S-02-2245. 69p. Dep. NTIS, PC A04/MF A01.

A physical model is proposed to derive the sweeping flow for LMFBR triangular array wire-wrapped assemblies under the turbulent flow condition. Two correlations are suggested for the sweeping flow through two different types of gaps between subchannels, the gap between the interior subchannels and the gap between the wall subchannels. These two sweeping flow correlations are evolved by calibrating the constants in the proposed model against the available experimental data. Agreement between the correlations and all the experimental data of $\pm 35\%$ is obtained over the assembly design range of $1.315 > P/D > 1.067$ and $52 > H/D > 4$. Based on these correlations, flow sweeping input parameters for two popular computer codes, i.e., COBRA and SUPERENERGY, are recommended.

888 (COO--2245-56TR) FLOW SPLIT MODEL FOR LMFBR WIRE WRAPPED ASSEMBLIES. Chiu, C.; Rohsenow, W.M.; Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA)). Apr 1978. Contract EY-76-S-02-2245. 49p. Dep. NTIS (US Sales Only), PC A03/MF A01.

A physical model is proposed to describe the subchannel pressure drop and in turn to derive two correlations for the interior subchannel and edge subchannel average flow rates in wire wrapped LMFBR assemblies. This model takes into account the effects of both the form drag induced by the wires and the skin friction pressure drop induced by the secondary flow in the determination of the interior and edge average subchannel flow rates. As a result, good agreement ($\pm 4\%$) between the predicted and the experimental subchannel flow rates is obtained for the bundles having the geometric characteristics ranging from $P/D=1.067$ and $H/D=4.0$ to $P/D=1.25$ and $H/D=52$ over the Reynolds number range $5000 < Re < 73,000$. The corner subchannel flow rates are not predicted by the proposed model because experimental data is lacking on the transverse and axial flow in the corner subchannels. However, when this information is available, the proposed model can be expanded to predict the corner subchannel flow rate.

890 (COO--2245-59TR) TWO DIMENSIONAL STRUCTURAL ANALYSIS OF REACTOR FUEL ELEMENT CLADDINGS DUE TO LOCAL EFFECTS. Karimi, R.; Wolf, L. (Massachusetts Inst. of Tech., Cambridge (USA)). Dept. of Nuclear Engineering). Apr 1978. Contract EY-76-C-02-2245. 431p. Dep. NTIS, MF A01.

Portions of document are illegible.

Two dimensional thermoelastic and inelastic stresses and deformation of typical LWR (PWR) and LMFBR (CRBR) claddings are evaluated by utilizing the following codes, for (1) Thermoelastic analysis (a) STRESS Code (b) SEPIPE Code (2) Thermoelastic analysis (a) Modified version of the GOGD code (b) One dimensional GRO-II code. The primary objective of this study is to analyze the effect of various local perturbations in the clad temperature field, namely eccentrically mounted fuel pellet, clad ovality, power tilt across the fuel and clad-coolant heat transfer variation on the cladding stress and deformation. In view of the fact that the thermoelastic analysis is always the first logical choice entering the structural field, it was decided to start the analysis with the two dimensional codes such as STRESS and SEPIPE. Later, in order to assess the validity and compare the thermoelastic results to those obtained for actual reactor conditions, a two dimensional code, namely a modified version of the GOGD code, was used to account for inelastic effects such as irradiation and thermal creep and swelling in the evaluation. The comparison of thermoelastic and inelastic results shows that the former can be used effectively to analyze LWR fuel pin over 350 hours of lifetime under the most adverse condition and 500 hours of lifetime for an LMFBR fuel pin. Beyond that the inelastic solution must be used. The impact of the individual thermal perturbation and combinations thereof upon the structural quantity is also shown. Finally, the effect of rod displacement on the two dimensional thermal and structural quantities of the LMFBR fuel pin cladding is analyzed.

891 (COO--2245-60) COOLANT MIXING IN LMFBR ROD BUNDLES AND OUTLET PLENUM MIXING TRANSIENTS. PROGRESS REPORT, SEPTEMBER 1, 1977--NOVEMBER 30, 1977. Todreas, N.E.; Golay, M.W.; Wolf, L. (Massachusetts Inst. of Tech., Cambridge (USA)). Dept. of Nuclear Engineering). 1977. Contract EY-76-S-02-2245. 53p. Dep. NTIS, PC A04/MF A01.

Progress in studies on LMFBR hydrodynamics is summarized under the following task headings:

889 (COO--2245-58TR) 3-D SLUG FLOW HEAT

bundle geometry (wrapped and bare rods), subchannel geometry (bare rods) outlet plenum flow mixing, and theoretical determination of local temperature fields in fuel rod bundles.

- 892 (COO--2245-61TR) MEASUREMENT OF HEAT AND MOMENTUM EDDY DIFFUSIVITIES IN RECIRCULATING LMFBR OUTLET PLENUM FLOWS. Manno, V.P.; Golay, M.W. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). Jun 1978. Contract EY-76-S-02-2245. 308p. Dep. NTIS, PC A14/MF A01.

An optical technique has been developed for the measurement of the eddy diffusivity of heat in a transparent flowing medium. The method uses a combination of two established measurement tools: a Mach-Zehnder interferometer for the monitoring of turbulently fluctuating temperature and a Laser Doppler Anemometer (LDA) for the measurement of turbulent velocity fluctuations. The technique is applied to the investigation of flow fields characteristic of the LMFBR outlet plenum. The study is accomplished using air as the working fluid in a small scale Plexiglas test section. Flows are introduced into both the $1/15$ scale FFTF outlet plenum and the $3/80$ scale CRBR geometry plenum at inlet Reynolds numbers of 22,000. Measurements of the eddy diffusivity of heat and the eddy diffusivity of momentum are performed at a total of 11 measurement stations. Significant differences of the turbulence parameters are found between the two geometries, and the higher chimney structure of the CRBR case is found to be the major cause of the distinction. Spectral intensity studies of the fluctuating electronic analog signals of velocity and temperature are also performed. Error analysis of the overall technique indicates an experimental error of 10% in the determination of the eddy diffusivity of heat and 6% in the evaluation of turbulent momentum viscosity. In general it is seen that the turbulence in the cases observed is not isotropic, and use of isotropic turbulent heat and momentum diffusivities in transport modelling would not be a valid procedure.

- 893 (COO--2245-62TR) LABORATORY MANUAL FOR SALT MIXING TEST IN ROD BUNDLES. Khan, H.U.R.; Chiu, C.; Todreas, N. (Massachusetts Inst. of Tech., Cambridge (USA)). Oct 1978. Contract EY-76-S-02-2245. 37p. Dep. NTIS, PC A03/MF A01.

This report is a Laboratory Manual dealing with the procedure employed during Salt Tracer Experiments, which are used for evaluating the hydraulic characteristics of a rod bundle. A description of the standard equipment used is given together with details of manufacture of non-standard items i.e., probes used for detecting the salt-concentration. Details of bundle construction have not been included as they are available in the references cited. An attempt has also been made to point out potential trouble areas and procedures.

- 894 (COO--2245-63) COOLANT MIXING IN LMFBR ROD BUNDLES AND OUTLET PLENUM MIXING TRANSIENTS. PROGRESS REPORT, DECEMBER 1, 1977--FEBRUARY 28, 1978. Todreas, N.E.; Golay, M.W.; Wolf, L. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 1978. Contract EY-76-S-02-2245. 46p. Dep. NTIS, PC A03/MF A01.

Progress is reported for the following tasks: bundle geometry (wrapped and bare rods), subchannel geometry (bare rods), velocity and temperature fields in outlet plenum flow mixing, and theoretical determination of local temperature fields in fuel rod bundles.

- 895 (COO--2245-64) COOLANT MIXING IN LMFBR ROD BUNDLES AND OUTLET PLENUM MIXING TRANSIENTS. PROGRESS REPORT, MARCH 1, 1978--MAY 31, 1978. Todreas, N.E.; Golay, M.W.; Wolf, L. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 1978. Contract EY-76-S-02-2245. 37p. Dep. NTIS, PC A03/MF A01.

An optical technique has been developed for the measurement of the eddy diffusivity of heat in a transparent flowing medium. The method uses a combination of two established measurement tools: a Mach-Zehnder interferometer for the monitoring of turbulently fluctuating temperature and a Laser Doppler Anemometer (LDA) for the measurement of turbulent velocity fluctuations. The technique is applied to the investigation of flow fields characteristic of the LMFBR outlet plenum. The study is accomplished using air as the working fluid in a small scale Plexiglas test section. Flows are introduced into both the $1/15$ scale FFTF outlet plenum and the $3/80$ scale CRBR geometry plenum at inlet Reynolds numbers of 22,000.

- 896 (COO--2245-65) COOLANT MIXING IN LMFBR ROD BUNDLES AND OUTLET PLENUM MIXING TRANSIENTS. PROGRESS REPORT, JUNE 1, 1978--AUGUST 31, 1978. Todreas, N.E.; Golay, M.W.; Wolf, L. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 1978. Contract EY-76-S-02-2245. 29p. Dep. NTIS, PC A03/MF A01.

Information is presented concerning coolant mixing in LMFBR fuel rods in wrapped and bare bundle geometry; LMFBR outlet plenum flow mixing; and theoretical determination of local temperature fields in LMFBR fuel rod bundles.

- 897 (COO--2245-66) COOLANT MIXING IN LMFBR ROD BUNDLES AND OUTLET PLENUM MIXING TRANSIENTS. PROGRESS REPORT, SEPTEMBER 1, 1978--NOVEMBER 30, 1978. Todreas, N.E.; Golay, M.W.; Wolf, L. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 1978. Contract EY-76-S-02-2245. 31p. Dep. NTIS, PC A02/MF A01.

Information is presented concerning coolant mixing in LMFBR fuel rods in wrapped and bare bundle geometry; LMFBR outlet plenum flow mixing; and theoretical determination of local temperature fields in LMFBR fuel rod bundles.

- 898 (COO--2245-68TR) MULTICELL SLUG FLOW HEAT TRANSFER ANALYSIS OF FINITE LMFBR BUNDLES. Yeung, M.K.; Wolf, L. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). Dec 1978. Contract EY-76-S-02-2245. 51p. Dep. NTIS, PC A22/MF A01.

An analytical two-dimensional, multi-region, multi-cell technique has been developed for the thermal analysis of LMFBR rod bundles. Local temperature fields of various unit cells were obtained for 7, 19, and 37-rod bundles of different geometries and power distributions. The validity of the technique has been verified by its excellent agreement with the THTB calculational result. By comparing the calculated fully-developed circumferential clad temperature distribution with those of the experimental measurements, an axial correction factor has been derived to account for the entrance effect for practical considerations. Moreover, the knowledge of the local temperature field of the rod bundle leads to the determination of the effective mixing lengths $L_{sub ij}$ for adjacent subchannels of various geometries. It was shown that the

implementation of the accurately determined L/sub i/ into COBRA-IIIC calculations has fairly significant effects on intersubchannel mixing. In addition, a scheme has been proposed to couple the 2-D distributed and lumped parameter calculation by COBRA-IIIC such that the entrance effect can be implanted into the distributed parameter analysis. The technique has demonstrated its applicability for a 7-rod bundle and the results of calculation were compared to those of three-dimensional analyses and experimental measurements.

- 899 (COO--2245-69) COOLANT MIXING IN LMFBR ROD BUNDLES AND OUTLET PLENUM MIXING TRANSIENTS. PROGRESS REPORT, DECEMBER 1, 1978--FEBRUARY 28, 1979. Todreas, N.E.; Golay, M.W.; Wolf, L. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 1979. Contract EY-76-S-02-2245. 35p. Dep. NTIS, PC A03/MF A01.

Information is presented concerning coolant mixing in bundle geometry and subchannel geometry; outlet plenum flow mixing; and theoretical determination of local temperature fields in LMFBR fuel rod bundles.

- 900 (COO--2245-70) COOLANT MIXING IN LMFBR ROD BUNDLES AND OUTLET PLENUM MIXING TRANSIENTS. PROGRESS REPORT, MARCH 1-MAY 31, 1979. Todreas, N.E.; Golay, M.W.; Wolf, L. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 1979. Contract EY-76-S-02-2245. 68p. Dep. NTIS, PC A04/MF A01.

Progress is summarized concerning liquid flow in bundle geometry (wrapped and bare rods) and LMFBR outlet plenum flow mixing.

- 901 (COO--2250-28) MIT LMFBR BLANKET PHYSICS PROJECT PROGRESS REPORT NO. 7, JULY 1, 1975--SEPTEMBER 30, 1976. Driscoll, M.J. (ed.). (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 30 Sep 1976. Contract EY-76-S-02-2250. 186p. (MITNE--211). Dep. NTIS, PC A09/MF A01.

Work during the period was devoted primarily to a range of analytical/numerical investigations, including evaluation of means to improve external blanket designs, beneficial attributes of the use of internal blankets, improved methods for the calculation of heterogeneous self-shielding and parametric studies of calculated spectral indices. Experimental work included measurements of the ratio of U-238 captures to U-235 fissions in a standard blanket mockup, and completion of development work on the radiophotoluminescent readout of LIF thermoluminescent detectors. The most significant findings were that there is very little prospect for substantial improvement in the breeding performance of external blankets, but internal blankets continue to show promise, particularly if they are used in such a way as to increase the volume fraction of fuel inside the core envelope. An improved equivalence theorem was developed which may allow use of fast reactor methods to calculate heterogeneously self-shielded cross sections in both fast and thermal reactors.

- 902 (COO--2250-29) MIT LMFBR BLANKET RESEARCH PROJECT. QUARTERLY PROGRESS REPORT, APRIL 1, 1977--JUNE 30, 1977. Driscoll, M.J. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 1977. Contract EY-76-S-02-2250. 13p. Dep. NTIS, PC A02/MF A01.

Axial power shaping calculations for the radial blanket of an LMFBR type reactor are presented.

- 903 (COO--2250-30) MIT LMFBR BLANKET RESEARCH PROJECT. QUARTERLY PROGRESS REPORT, JULY 1, 1977--SEPTEMBER 30, 1977. Driscoll, M.J. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 1977. Contract EY-76-S-02-2250. 16p. Dep. NTIS, PC A02/MF A01.

Irradiation experiments on Mockup 5B have been completed. A series of comparative calculations are underway on internal blanket variations. Preliminary results are presented. A final workup of the T/sub i/(n,p) Sc⁴⁸ and N/sub i/(n,p) Co⁵⁸ threshold traverses in Blanket Mockup 5B has been completed.

- 904 (COO--2250-31) MIT LMFBR BLANKET RESEARCH PROJECT. QUARTERLY PROGRESS REPORT, OCTOBER 1, 1977--DECEMBER 31, 1977. Driscoll, M.J. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 1977. Contract EY-76-S-02-2250. 16p. Dep. NTIS, PC A02/MF A01.

Research progress on LMFBR breeding blanket is reported for the following: threshold foil traverses in blanket/reflector Mockup 5B, Mockup No. 6 assemblies, and LCCEWG benchmark problems.

- 905 (COO--2250-32) MIT LMFBR BLANKET RESEARCH PROJECT. QUARTERLY PROGRESS REPORT, JANUARY 1, 1978--MARCH 31, 1978. Driscoll, M.J. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 1978. Contract EY-76-S-02-2250. 15p. Dep. NTIS, PC A02/MF A01.

Progress in the development of LMFBR breeding blanket design parameters is reported under the following headings: thorium utilization, reflector fast neutron penetration, and interface measurement and calculations.

- 906 (COO--2250-34) MIT LMFBR BLANKET RESEARCH PROJECT. QUARTERLY PROGRESS REPORT, JULY 1, 1978--SEPTEMBER 30, 1978. Driscoll, M.J. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 20 Oct 1978. Contract EY-76-S-02-2250. 9p. Dep. NTIS, PC A02/MF A01.

Progress in the development of LMFBR breeding blanket design parameters is reported under the following headings: an improved Dancoff factor prescription and breed/burn blanket management.

- 907 (COO--2250-36) MIT LMFBR BLANKET RESEARCH PROJECT. QUARTERLY PROGRESS REPORT, JANUARY 1, 1979--MARCH 31, 1979. Driscoll, M.J. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 20 Apr 1979. Contract EY-76-S-02-2250. 14p. Dep. NTIS, PC A02/MF A01.

Progress is summarized in the following areas: fissile build-up in internal blankets, reference cores, and power peaking.

- 908 (COO--2250-37) INTERFACIAL EFFECTS IN FAST REACTORS. Saidi, M.S.; Driscoll, M.J. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). May 1979. Contract EY-76-S-02-2250. 196p. Dep. NTIS, PC A09/MF A01.

The problem of increased resonance capture rates near zone interfaces in fast reactor

media has been examined both theoretically and experimentally. An interface traversing assembly was designed, constructed and employed to measure U-238 capture rates near the blanket-reflector interface in the MIT Blanket Test Facility. Prior MIT experiments on a thorium-uranium interface in a blanket assembly were also reanalyzed. Extremely localized fertile capture rate increases of on the order of 50% were measured immediately at the interfaces relative to extrapolation of asymptotic interior traverses, and relative to state-of-the-art (LIB-IV, SPHINX, ANISN/2DB) calculations which employ infinite-medium self-shielding throughout a given zone. A method was developed to compute a spatially varying background scattering cross section per absorber nucleus which takes into account both homogeneous and heterogeneous effects on the interface flux transient.

- 909 (COO--2250-38) MIT LMFBR BLANKET RESEARCH PROJECT. QUARTERLY PROGRESS REPORT, APRIL 1-JUNE 30, 1979. Driscoll, M.J. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). 25 Jul 1979. Contract EY-76-S-02-2250. 16p. Dep. NTIS, PC A02/MF A01.

In designing a plutonium-uranium core with thorium internal blankets there are several parameters that must be investigated to identify the design with the best performance. Choices include: the amount of moderation in the blanket, whether moderation is incorporated from the outset or added after an assembly is shuffled to the external blanket, and the cycle length both for the core and internal blanket assemblies. To investigate the effect of these parameters several core and blanket burnup analyses were performed.

- 910 (COO--2426-101) SYSTEM DESIGN OF CARBIDE FUEL. QUARTERLY PROGRESS REPORT, APRIL--JUNE 1977. Noyes, R.C. (Combustion Engineering, Inc., Windsor, CT (USA)). 1977. Contract EY-76-C-02-2426. 76p. (CE-FBR--77-339). AT.

The expected costs and benefits of introducing carbide fueled assemblies into Liquid Metal Fast Breeder Reactors are studied. Associated with this overall objective are the parallel objectives of identifying the important characteristics of optimum commercial carbide fueled LMFBR designs and the comparison of such designs with designs of oxide fueled LMFBR's. Determination of efficient and acceptable means of testing carbide and other advanced fuels in FTR and CRBR, including detailed design and analysis of experimental FTR driver fuel assemblies, is a major objective of this work.

- 911 (COO--2426-133) PERFORMANCE POTENTIAL OF (TH,U) CARBIDE AND (TH,U) NITRIDE FUEL IN 1200 MWE LMFBR'S. Caspersson, S.A.; Kulwich, M.R. (Combustion Engineering, Inc., Windsor, CT (USA)). Apr 1978. Contract EY-76-C-02-2426-001. 10p. Dep. NTIS, PC A02/MF A01.
- An evaluation of the performance potential of thorium-uranium carbide and nitride fuel in 1200 MW(e) homogeneous and heterogeneous LMFBR's has been completed. Comparisons were done with 9.40 mm outer diameter, 0.38 mm cladding fuel pins at a selected peak (3 sigma) linear power of 98.4 kW/m for all cores. Breeding ratios for the homogeneous (Th,U)C and (Th,U)N cores are 1.12 and 1.16, respectively. In the heterogeneous cores the breeding ratios are 0.08 to 0.09 greater. A comparative breeding ratio of 1.48 results for a sodium-bonded (U,Pu)C homogeneous LMFBR with similar

design criteria. Compound inventory doubling times for the homogeneous and heterogeneous (Th,U)C cores are 82 and 53 years, respectively. Additional performance parameters have also been compared, including sodium void worths, specific inventories, fissile material gains, and equilibrium cycle reactivity losses.

- 912 (COO--2426-136) SYSTEM DESIGN OF CARBIDE FUEL. QUARTERLY PROGRESS REPORT, JANUARY--MARCH 1978. (Combustion Engineering, Inc., Windsor, CT (USA)). 1978. Contract EY-76-C-02-2426. 30p. (CE-FBR--7878). AT.
- Portions of document are illegible.
- Progress is summarized in design studies and performance testing of advanced carbide fuels for LMFBR reactors. (DG)

- 913 (COO--2458-22) SIMPLE REPRESENTATIONS OF SPECTRUM AVERAGED CROSS SECTIONS IN LMFBR BLANKETS. Badruzzaman, A.; Wiley, R.; Becker, M. (Rensselaer Polytechnic Inst., Troy, NY (USA). Dept. of Nuclear Engineering). 1978. Contract EY-76-S-02-2458. 7p. (CONF-781105-101). Dep. NTIS, PC A02/MF A01.
- From ANS meeting; Washington, DC, USA (12 Nov 1978).

The setting up of correlations to represent spectrum averaged cross sections in the core-blanket interface and blanket region of LMFBRs is described. Analytical representations of concentrations as functions of position are used. Sample calculations using a CRBR core composition are presented.

- 914 (COO--2458-23) ANALYSIS OF EXPOSURE DEPENDENT HETEROGENEOUS FBR SPECTRA USING CONTINUOUS SLOWING DOWN-INTEGRAL TRANSPORT CONCEPTS. Badruzzaman, A.; Badruzzaman, T.; Becker, M. (Rensselaer Polytechnic Inst., Troy, NY (USA). Dept. of Nuclear Engineering). 1979. Contract EY-76-S-02-2458. 7p. (CONF-790602--44). Dep. NTIS, PC A02/MF A01.
- From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

Generalized continuous slowing down (CSD) theory coupled with integral transport concepts has been shown to give accurate prediction of the spatial variation of exposure dependent spectra at the core-blanket interface and in the blanket region of the homogeneous design fast reactor. The spectrum and the spectrum-averaged cross sections (SAXS) exhibited a strong space dependence as one proceeded from the outer core into the blanket. However, at any given location these varied rather weakly with exposure. In the paper the authors have extended the procedure to analyse the heterogeneous LMFBR spectra with depletion. Briefly, one solves a matrix differential equation of the zonal slowing down densities; the equation being a generalization of slowing down equations of the Greuling--Goertzel type with a source term. One then uses a multiregion analogue of the usual expressions of flux in terms of slowing down densities, either of the Greuling--Goertzel or Wigner form. The formalism has been applied to an infinite array of alternating core and blanket zones in slab geometry. The properties of these zones are based on Clinch River Breeder Reactor (CRBR) parameters and calculations were performed with the 29 group FTR-300 data set.

- 915 (COO--2458-24) SIMPLE PROCEDURE FOR DETERMINING IMPLICATIONS OF DESIGN CHANGES ON FAST REACTOR FUEL CYCLE COST AND BREEDING PERFORMANCE. Badruzzaman, A.; Becker, M. (Rensselaer Polytechnic Inst., Troy, NY (USA). Dept. of Nuclear Engineering). 1979.

Contract EY-76-S-02-2458. 6p. (CONF-790602--18). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

Recently, analytical sensitivity methods has been applied to obtain fuel cycle cost implications of data uncertainties. To perform exposure dependent sensitivity analysis without repeating expensive spectrum calculations, simple correlations of the spectrum averaged cross sections (SAXS) were constructed. The correlation coefficients were obtained by fitting the SAXS calculated by direct methods over a wide range of LMFBR core designs. In this paper the procedure has been extended to study sensitivity of fuel cycle cost and breeding ratio to design variation. The method involves using the correlations to construct both the SAXS and the sensitivity coefficients. Composition dependent correlations have been found to be accurate for the core while both composition and position have to be included in analysing the blanket.

- 916 (COO--2509-4) MECHANISMS OF FAST NEUTRON PENETRATION IN THICK LAYERS OF SODIUM. Huang, L.Y. (Columbia Univ., New York (USA). Div. of Nuclear Science and Engineering). 1975. Contract EY-76-S-02-2509. 219p. Dep. NTIS, PC A10/MF A01.

Thesis.

A series of computer experiments was carried out to elucidate the penetration mechanisms of fast neutrons through thick layers of sodium such as occur in LMFBR designs. As a one-dimensional approximation of the actual situation, the calculations concentrated mainly on the flux 5 meters from a plane isotropic fission source in an infinite sodium medium. Most of the transport calculations were made with the moments-method code BMT with a 496-energy point grid. Previously developed methods for reconstructing the flux from the spatial moments were used, except that a set of biorthogonal polynomials was constructed suitable for expansion of the flux in terms of a Gaussian weight function. The moments-method technique lends itself to easy and economical changes of the input cross section data. A large number of such modified cross section sets, built around the ENDF/B-III set, were used in separate calculations designed variously to emphasize or eliminate one or more particular transport processes. It was shown that, as the energy decreases below 190 keV, the flux spectrum at 5 m is increasingly dominated by an age-diffusion process that is quantitatively close to conventional age theory if the age is suitably chosen. Conclusions from this picture of neutron penetration in sodium are made as to the types of transport calculations that can be successfully made in shield design, and the accuracies needed in future cross section measurements. 37 figures, 30 tables.

- 917 (COO--2554-6) SIMULATION OF BOILING POOLS WITH INTERNAL HEAT SOURCES BY GAS INJECTION. Luk, A.C.H.; Ganguli, A.; Bankoff, S.G. (Northwestern Univ., Evanston, IL (USA). Dept. of Chemical Engineering). 29 Jul 1977. Contract EY-76-S-02-2554. 32p. Dep. NTIS, PC A03/MF A01.

Heat transfer from the sides and bottom of an open non-boiling liquid pool with spatially uniform internal gas injection was studied experimentally, both in transient and in steady state. The results were compared with experimental data for a boiling pool without permanent gas injection undertaken at Argonne National Laboratory. Typical Nusselt number versus Reynolds number plots showed that heat

transfer rates were much higher in the gassy pool due to more efficient circulation. A correction was applied to estimate the surface evaporation effect under boiling conditions. Bubble size and distribution effects controlled the heat transfer rates. Vertical void fraction profiles were inferred from local static pressure measurements. A modified Grashof number, in terms of the average void fraction, was also used to correlate the data for horizontal heat flow. The bottom and side heat loss rates were about equal, which would indicate that the pool might retain its shape as it sinks into the support material.

- 918 (COO--2571-9) IMPROVEMENT AND VERIFICATION OF FAST REACTOR SAFETY ANALYSIS TECHNIQUES. PROGRESS REPORT, APRIL 1, 1977--JUNE 30, 1977. Barker, D.H.; Wheeler, P.A. (Brigham Young Univ., Provo, UT (USA). Dept. of Chemical Engineering). 1977. Contract EY-76-S-02-2571. 17p. Dep. NTIS, PC A02/MF A01.

An energy balance was made to determine the effect of the latent heat of vaporization of fuel and steel on temperature changes of the fuel and steel. Calorimeter runs were made on the reaction of DMSO with acetyl chloride to determine the effects of concentration and injection speed on the power of the reaction. Within a limited range, the power is proportional to both the concentration and injection rate. A plot of void fraction as a function of concentration was made. The concentration at which the reaction suddenly produces a large void fraction is between 3 and 4 molar.

- 919 (COO--2571-10) IMPROVEMENT AND VERIFICATION OF FAST REACTOR SAFETY ANALYSIS TECHNIQUES. PROGRESS REPORT, 1 JULY 1977--30 SEPTEMBER 1977. Barker, D.H.; Wheeler, P.A. (Brigham Young Univ., Provo, UT (USA). Dept. of Chemical Engineering). 1977. Contract EY-76-S-02-2571. 30p. Dep. NTIS, PC A02/MF A01.

A calorimeter was designed and built to measure reaction power in the range where liquid is boiling. Measurements made with the new calorimeter at a concentration just lower than what would produce boiling (3m) was about 10% lower than those measured using the Precision Calorimeter. A heat loss of about 40% due to a non-adiabatic vessel was determined by comparing the reaction power in solution of the reaction vessel to that of the calorimeter. A closed reaction vessel was designed to observe the effects of pressure (up to 3 atm) on the dispersion height. A preliminary calculation of the theoretical critical superficial vapor velocity was 0.0432 m/sec.

- 920 (COO--2571-11) IMPROVEMENT AND VERIFICATION OF FAST REACTOR SAFETY ANALYSIS TECHNIQUES. PROGRESS REPORT, OCTOBER 1, 1977--DECEMBER 31, 1977. Barker, D.H.; Wheeler, P.A. (Brigham Young Univ., Provo, UT (USA). Dept. of Chemical Engineering). 1977. Contract EY-76-S-02-2571. 38p. Dep. NTIS, PC A02/MF A01.

The critical superficial vapor velocity was measured for an open vessel system with chemically produced internal heat. The measured value was within experimental error of the value predicted by the Kutateladze stability criterion. The greatest error induced was in the measurement of the energy generation rate. The effects of mixtures were taken into account. Further work suggestions are also presented.

- 921 (COO--2571-12) IMPROVEMENT AND

VERIFICATION OF FAST REACTOR SAFETY ANALYSIS TECHNIQUES. PROGRESS REPORT, JANUARY 1, 1978--MARCH 31, 1978. Barker, D.H.; Wheeler, P.A. (Brigham Young Univ., Provo, UT (USA). Dept. of Chemical Engineering). 1978. Contract EY-76-S-02-2571. 11p. Dep. NTIS, PC A02/MF A01.

A stainless steel vessel was used to perform preliminary fuel simulant, closed system experiments. The reaction of DMSO in acetyl chloride was carried out within the closed vessel. Temperatures and pressure in the vessel were measured as a function of time for several solution concentrations. These experiments are the first of a series designed to investigate the properties of fuel dispersion in a closed system.

- 922 (COO--2571-14) IMPROVEMENT AND VERIFICATION OF FAST REACTOR SAFETY ANALYSIS TECHNIQUES. PROGRESS REPORT, JULY 1, 1978--SEPTEMBER 30, 1978. Barker, D.H.; Wheeler, P.; Bybee, D.J. (Brigham Young Univ., Provo, UT (USA). Dept. of Chemical Engineering). 1978. Contract EY-76-S-02-2571. 169p. Dep. NTIS, PC A08/MF A01.

This report deals with the simulation of a HCDA in a LMFBR using chemical heating. The chemicals used were acetyl chloride and dimethyl sulfoxide in a benzene base. In our last report, it was shown that for high concentrations of the two chemicals, vapor velocities above 8 cm/sec and void fractions above 50% were obtained in a closed system. However, at lower concentrations, such as 4 M, no void fraction was obtained. During this period of work, a closed system with a cooling coil was used to provide a larger heat sink. It was shown that for the 4 M concentration void fractions above 50% was obtained and for the 9 M concentration the void fraction was higher and stayed dispersed much longer than without the cooling coil.

- 923 (COO--2571-15) IMPROVEMENT AND VERIFICATION OF FAST REACTOR SAFETY ANALYSIS TECHNIQUES. PROGRESS REPORT, OCTOBER 1, 1978--DECEMBER 31, 1978. Barker, D.H.; Wiberg, D.V. (Brigham Young Univ., Provo, UT (USA). Dept. of Chemical Engineering). 1978. Contract EY-76-S-02-2571. 11p. Dep. NTIS, PC A02/MF A01.

The effect of using Acetone, Benzene, or Toluene as a solvent material in a 4M, Acetyl Chloride-Dimethyl Sulfoxide reaction system was investigated. Maximum void fraction and length of time with void greater than 50% were compared. The Acetone system produced the largest maximum void and longest time above 50% void followed by Benzene and Toluene in descending order. Seventeen waxes were tested and compared to determine a material suitable to simulate reactor containment materials (e.g., steel, concrete). Waxes were compared on the basis of solubility in the reaction materials, melting point and structural integrity.

- 924 (COO--2571-17) IMPROVEMENT AND VERIFICATION OF FAST REACTOR SAFETY ANALYSIS TECHNIQUES. PROGRESS REPORT, APRIL 1, 1979--JUNE 30, 1979. Barker, D.H.; Wiberg, D.V. (Brigham Young Univ., Provo, UT (USA). Dept. of Chemical Engineering). 1979. Contract EY-76-S-02-2571. 36p. Dep. NTIS, PC A03/MF A01.

This report details a mathematical model for the dissipation of heat from an internally heated liquid pool which is contained in a hollow cylinder of finite inside and outside diameter. The model includes a temperature profile for the previously described cylinder, empirical correlations for Nusselt numbers at the walls of the container and a film theory

correction of the heat transfer coefficients for mass transfer. Also included in this report is a description of the experimental work conducted as a preliminary step to verification or rejection of the proposed mathematical model. This experimental work consisted of reacting Dimethyl Sulfoxide (DMSO) with Acetyl Chloride (AC), using Acetone as a solvent, in a hollow wax cylinder and observing the geometry and dimensions of the melt.

- 925 (COO--2571-19) IMPROVEMENT AND VERIFICATION OF FAST REACTOR SAFETY ANALYSIS TECHNIQUES. PROGRESS REPORT, OCTOBER 1, 1979--MARCH 31, 1980. Barker, D.H.; Wiberg, D.V. (Brigham Young Univ., Provo, UT (USA). Dept. of Chemical Engineering). 1980. Contract AC02-76ET37205. 7p. NTIS, PC A02/MF A01.

Several of the time dependent properties of the dimethyl sulfoxide-acetyl chloride reaction system were determined. The volume of a wax cylinder melted by conducting a dimethyl sulfoxide-acetyl chloride reaction in the vessel was compared with the melted volume calculated using several different heat transfer correlations. It was found that a laminar flow at the wall model provided the most accurate calculation of the observed melted volume.

- 926 (COO--2781-12TR) HEAT TRANSFER AND FLUID FLOW ASPECTS OF FUEL--COOLANT INTERACTIONS. Corradini, M.L. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). Sep 1978. Contract EY-76-S-02-2781. 530p. Dep. NTIS, PC A23/MF A01.

A major portion of the safety analysis effort for the LMFBR is involved in assessing the consequences of a Hypothetical Core Disruptive Accident (HCDA). The thermal interaction of the hot fuel and the sodium coolant during the HCDA is investigated in two areas. A postulated loss of flow transient may produce a two-phase fuel at high pressures. The thermal interaction phenomena between fuel and coolant as the fuel is ejected into the upper plenum are investigated. A postulated transient overpower accident may produce molten fuel being released into sodium coolant in the core region. An energetic coolant vapor explosion for these reactor materials does not seem likely. However, experiments using other materials (e.g., Freon/water, tin/water) have demonstrated the possibility of this phenomenon.

- 927 (COO--2781-13) FUEL COOLANT THERMAL INTERACTION PROJECT UC 79P. QUARTERLY PROGRESS REPORT NO. 7. (Massachusetts Inst. of Tech., Cambridge (USA)). Jul 1977. Contract EY-76-S-02-2781. 20p. Dep. NTIS, PC A02/MF A01.

The objective of the continued work on this project is to experimentally and analytically study the dominant mechanisms in fuel-coolant thermal interactions which could lead to vapor explosions. The exploration of mechanisms is focused in two areas: (1) investigations in dynamics of liquid-liquid film boiling; and (2) possible mechanisms of energy dissipation due to heat transfer during the HCDA expansion. Work being performed simultaneously in these areas is briefly described.

- 928 (COO--2781-14) FUEL--COOLANT THERMAL INTERACTION PROJECT. PROGRESS REPORT NO. 8. Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA)). 1978. Contract EY-76-S-02-2781. 24p. Dep. NTIS, PC A02/MF A01.

Progress is reported in the areas of: (a) Analysis of the experimental results of a fundamental Taylor Instability Experiment; (b) Development of simple models for liquid entrainment and liquid-vapor heat transfer occurring in the transient expansion of a gaseous high pressure zone; and (c) Application of these models experiments to assess the model's adequacy.

- 929 (COO--2781-15FR) FINAL REPORT ON MIT CONTRACT ON FUEL-COOLANT INTERACTIONS. Todreas, N.E. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). Oct 1978. Contract EY-76-S-02-2781. 20p. Dep. NTIS, PC A02/MF A01.

The work of this contract has yielded results in three areas which can be summarized as follows. The existing body of molten metal/water interaction data appears consistent with the criteria of the spontaneous nucleation theory. The observed fragmentation of UO_2 can be satisfactorily predicted using the proposed thermal stress initiated fracture model. The work potential of a hypothetical fuel vapor source acting on a sodium slug can be reduced by about a factor of two by considering entrainment due to Taylor instabilities at the vapor-liquid interface.

- 930 (COO--2801-7) ACOUSTIC AND TURBULENT AGGLOMERATION OF SODIUM AEROSOLS. PROGRESS REPORT, JANUARY 1, 1977--DECEMBER 31, 1977. Hinds, W.C. (Harvard Univ., Boston, MA (USA). School of Public Health). Sep 1977. Contract EY-76-S-02-2801. 8p. Dep. NTIS, PC A02/MF A01.

Freshly formed and aged sodium aerosols generated by burning metallic sodium in air were decontaminated in wet cell washers of experimental design. Two wetted cells in series packed with curly glass fibers 35 μ m in diameter gave removal efficiencies in excess of 90%. More efficient cell packings and a larger number of cells in series can be used to raise collection efficiency above 99%.

- 931 (COO--2801-8) ACOUSTIC AND TURBULENT AGGLOMERATION OF SODIUM AEROSOLS. QUARTERLY PROGRESS REPORT, OCTOBER 1, 1977--DECEMBER 31, 1977. Hinds, W.C. (Harvard Univ., Boston, MA (USA). School of Public Health). Jan 1978. Contract EY-76-S-02-2801. 11p. Dep. NTIS, PC A02/MF A01.

Design of a wet cell washer and test system for air cleaning of sodium fire aerosols is described. The test aerosol was produced by continuous burning of metallic sodium in a 90 m^3 chamber to maintain a concentration of 1.14 ± 0.05 g/ m^3 for as long as 65 minutes. The sodium fire aerosol was drawn through the wet cell washer test loop at a constant flow rate of 9 m^3 /min. The wet cell washer consisted of two cells in series, the first, two off-set layers of perforated metal screen, and the second, a 4" thick filter stage packed with curly glass fibers 35 μ m in diameter. The wet cells were washed by solid cone water spray at 16.7 lpm at 30 psig. Sodium removal efficiencies ranged from 87 to 94%. Curly glass fibers were found to deteriorate after a six hour exposure to sodium aerosols.

- 932 (COO--2801-9) ACOUSTIC AND TURBULENT AGGLOMERATION OF SODIUM AEROSOLS. FINAL REPORT, OCTOBER 1, 1975--JULY 31, 1978. Hinds, W.; Price, J.; First, M.W. (Harvard Univ., Boston, MA (USA). School of Public Health). Nov 1978. Contract EY-76-S-02-2801. 15p. Dep. NTIS, PC A02/MF A01.

A suitability of 6 packing materials for removal of sodium fire aerosols in a wet cell scrubber was evaluated experimentally by measuring the efficiency and flow resistance of two 10.2 cm deep cells in series at air face velocities of 185 and 289 cm/sec, water flow rates of 0.210 and 0.387 $cm^3/sec/cm^2$, with and without aerosol prehumidification. The most satisfactory material was found to be 50 μ m diameter stainless steel fibers at a packing density of 30 kg/ m^3 (porosity = 0.99). Two 10.2 cm deep cells in series gave a removal efficiency of 85% for typical sodium fire aerosols at a face velocity of 289 cm/sec. Measurement of efficiency as a function of particle size indicated a reduction in efficiency in the 0.4 to 1.1 μ m size range for all packing materials.

- 933 (COO--2826-4) FAST BREEDER BLANKET FACILITY (FBBF). QUARTERLY PROGRESS REPORT, APRIL 1, 1977--JUNE 30, 1977. Ott, K.O. (ed.). (Purdue Univ., Lafayette, IN (USA)). Jun 1977. Contract EY-76-S-02-2826. 51p. (PNE--77-119). Dep. NTIS, PC A04/MF A01.

The work performed in the reporting period was primarily concerned with aspects of the preanalysis and the corresponding safety investigations of the Fast Breeder Blanket Facility (FBBF). Progress on the loading of the first configuration, testing experimental equipment, and performing check-out tests is also summarized. It is necessary that a neutron poison (B_4C) be placed in the converter regions in order to hold the value of k_{sub} eff/ for a hypothetical flooded FBBF of future loadings below the licensed limit of 0.75. It appears that this limit on k_{sub} eff/ can be attained for all loadings if one of the 180° configurations consists of stainless steel double clad natural uranium rods. The safety of one of the most reactive configurations is investigated and the need for B_4C verified. In addition, the safety of other FBBF loadings is investigated.

- 934 (COO--2826-5) FAST BREEDER BLANKET FACILITY (FBBF). QUARTERLY PROGRESS REPORT, JULY 1, 1977--SEPTEMBER 30, 1977. PNE-77-120. Ott, K.O. (ed.). (Purdue Univ., Lafayette, IN (USA). School of Nuclear Engineering). Sep 1977. Contract EY-76-S-02-2826. 60p. Dep. NTIS, PC A04/MF A01.

The work performed in the reporting period was primarily concerned with the detailed preanalysis of the first experimental loading sequence. Progress on testing experimental equipment and performing checkout test is also summarized. Preanalysis calculations of important reaction rates have been completed. Two-dimensional diffusion calculations were made to obtain the neutron flux. A program was written to obtain reaction rates at any spatial mesh point using the calculated flux. Preanalysis calculations for gamma-ray heating rates have also been completed. Coupled neutron-gamma-ray calculations were performed using a one-dimensional transport code to obtain the gamma-ray flux. Heating rates in several materials were then calculated using kerma factors. These preanalysis calculations of reaction rates and gamma-ray heating rates will be used in the detailed planning of the experiments.

- 935 (COO--2826-7) FAST BREEDER BLANKET FACILITY (FBBF). ANNUAL REPORT, JANUARY 31, 1976--DECEMBER 31, 1977. Ott, K.O. (ed.). (Purdue Univ., Lafayette, IN (USA). School of Nuclear Engineering). Jan 1978. Contract EY-76-S-02-2826. 85p. (PNE--78-127). Dep.

NTIS, PC A05/MF A01.

The work performed in the reporting period was primarily concerned with the construction of the Fast Breeder Blanket Facility (FBBF), acquisition of experimental equipment, outlining the experimental program, preanalysis of the initial loading configuration and investigation of the safety of the initial loading and advanced loadings. The detailed physical description of the FBBF, operational procedures and controls, radiation shielding and experimental equipment are presented. The ability of the FBBF to simulate the blanket spectrum of a large fast breeder reactor is illustrated by comparison of spectra. The source axial distribution, reaction rate comparisons, breeding of plutonium and gamma-ray energy deposition rates are also discussed. Some of the safety aspects of the initial loading and advanced loadings are described. Experimental capabilities of the facility are outlined.

- 936 (COO--2826-8) FAST BREEDER BLANKET FACILITY. QUARTERLY PROGRESS REPORT, JANUARY 1, 1978--MARCH 31, 1978. Clikeman, F.M. (ed.). (Purdue Univ., Lafayette, IN (USA). School of Nuclear Engineering). Mar 1978. Contract EY-76-S-02-2826. 39p. (PNE--78-128). Dep. NTIS, PC A03/MF A01.

The work performed in the reporting period was primarily concerned with the continued preanalysis of the first experimental sequence (Task B) and the loading of the first blanket (Task C). Progress on testing experimental equipment and performing checkout tests is also summarized. A comparison of calculated spectra for the original FBBF design, the "as-built" facility, and a large LMFBR is reported. A description of the one- and two-dimensional models used for reaction rate and spectrum calculations are summarized together with a comparison of the results.

- 937 (COO--2826-9) FAST BREEDER BLANKET FACILITY. QUARTERLY PROGRESS REPORT, APRIL 1, 1978--JUNE 30, 1978. Clikeman, F.M. (ed.). (Purdue Univ., Lafayette, IN (USA). School of Nuclear Engineering). Jun 1978. Contract EY-76-S-02-2826. 51p. (PNE--78-132). Dep. NTIS, PC A04/MF A01.

The work performed in the reporting period was primarily concerned with the initial measurements and checkout of the Purdue Fast Breeder Blanket Facility (TASK C). The safety calculations for the "as built" are also presented and compared with the initial safety calculations (TASK B). Safety calculations on the FBBF facility have been completed for a variety of new flooded-dry combinations. Specifically the effect of adding boron carbide to the transformer regions was investigated. The investigation showed the stabilizing effect of the boron carbide in the flooded cases while the effect in the dry cases is minimal. Measurements of the $k_{\text{sub eff}}$ of the FBBF facility, of the effectiveness of the shielding, and of the radiation levels in the rooms surrounding the laboratory have been completed and compare well with the calculations presented in the license application.

- 938 (COO--3027-27) EXPERIMENTAL HIGH TEMPERATURE COEFFICIENTS OF COMPRESSIBILITY AND EXPANSIVITY OF LIQUID SODIUM AND OTHER RELATED PROPERTIES. Das Gupta, S. (Columbia Univ., New York (USA)). 1977. Contract EY-76-S-02-3027. 172p. Dep. NTIS, MF A01.

Portions of document are illegible.

The subcooled compressibility of liquid

sodium was directly measured up to 200 atm between 900 K and 1867 K, utilizing a new multi-property apparatus which was previously tested with water. The experimental data were correlated by a 6-term equation with a standard deviation of 9.2 percent. The equation can be used to estimate the subcooled compressibilities and densities of liquid sodium up to 2300 K and 500 ata. The thermal expansion of liquid sodium was also measured along the isobars 1 ata, 28.9 ata and 69 ata. Densities within 1 percent of those obtained from the compressibilities were obtained. The above compressibility data were used to calculate the thermal pressure coefficient of saturated liquid sodium. Also, Bhise and Bonilla's correlations for the vapor pressure and the saturated liquid density of sodium were improved by including more data in the analysis. The critical temperature and density were thus reestimated as 2508.7 K and 0.2141 g/cc. Furthermore, a new correlation was developed to determine the heat of vaporization of sodium up to the critical point, which was then used to estimate the internal energy and the entropy of vaporization and the saturated vapor density of sodium up to the critical point.

- 939 (COO--3027-37) HIGH TEMPERATURE PROPERTIES OF NUCLEAR REACTOR COOLANTS AND THERMODYNAMIC POWER CYCLE WORKING FLUIDS. TECHNICAL PROGRESS REPORT, DECEMBER 1, 1977--NOVEMBER 30, 1978. Bonilla, C.F.; Holder, G.P. (Columbia Univ., New York (USA). Liquid Metals Research Lab.). Sep 1978. Contract EY-76-S-02-3027. 39p. Dep. NTIS, PC A03/MF A01.

The program to determine the surface tension of sodium to high temperatures has been soundly commenced with the completion and calibration of a new all-molybdenum maximum-bubble-pressure apparatus suitable to some 1900°K. This property bears on boiling, condensing and two-phase flow phenomena, also related to LMFBR analysis. This same apparatus will be useful with lithium, of interest in fusion reactor cooling. Reduction of prior maximum bubble pressure results on potassium by the Schroedinger method has been repeated by the more precise but time-consuming Sugden procedure, which has increased the resulting surface tension values by an average of 0.27%. Thus the Sugden method will be employed to reduce the data for sodium and lithium.

- 940 (COO--3536-13) SEPARATION OF KRYPTON AND XENON FROM REACTOR ATMOSPHERES BY SELECTIVE PERMEATION. FINAL REPORT. Stern, S.A.; Leone, S.M.; Onorato, F.J.; Wang, S.C. (Syracuse Univ., NY (USA). Dept. of Chemical Engineering and Materials Science). Mar 1978. Contract EY-76-S-02-3536. 110p. Dep. NTIS, PC A06/MF A01.

The removal of krypton and xenon from nuclear reactor atmospheres by permeation through silicone rubber capillaries has been studied theoretically and experimentally. Permeator modules resembling shell-and-tube heat exchangers have been constructed and tested. Permeation cascades were designed for the removal of krypton and xenon from the following nuclear atmospheres: (a) the atmosphere of a reactor containment building after a nuclear accident; (b) the off-gas of a nuclear fuel reprocessing plant; and (c) the cover gas of a liquid-metal-cooled fast breeder reactor. Design criteria are presented for the separation of multicomponent mixtures with large separation factors. The performance of "ideal" cascades, in which the concentrations of a key component are matched in mixing

interstage streams, is compared with that of cascades with constant stage cuts. The removal of krypton and xenon from reactor atmospheres in permeation cascades with silicone rubber capillaries is a potentially attractive process because of its safety and reliability. Permeator modules are compact and sturdy and a permeation plant could be easily transported to the scene of a nuclear accident.

- 941 (COO--4040-3) SURFACE RADIATION PROPERTIES BETWEEN NUCLEAR REACTOR MATERIALS IN CONTACT. Chan, S.H.; Tseng, H.H. (Wisconsin Univ., Milwaukee (USA). Dept. of Energetics). 18 Oct 1978. Contract EY-76-S-02-4040. 42p. Dep. NTIS, PC A03/MF A01.

This technical report summarizes a portion of the results on surface radiation properties of reactor materials, which is part of the program on Attenuation of Radiological Consequences From Core Disruptive Accidents by Radiative Transfer. In analyzing radiant energy exchange between the fuel and coolant, it is necessary to know the reflection absorption, and emission characteristics of the fuel and its interface with coolant. Because of unavailability and great uncertainty of those radiation properties, they are calculated from the electromagnetic theory of optics and results are presented here. Depending on the contacting medium, six types of surfaces received consideration. They are the interfaces between uranium dioxide and sodium, uranium dioxide and steel, uranium dioxide and a gas, water and a gas, steel and a gas as well as sodium and a gas. The spectral interface reflectance as well as the spectral absorptance from one side of the medium to the other side is evaluated for all wavelengths. Those spectral properties are further integrated to yield the total hemispherical properties for a black body source over a temperature range of 1200°K to 6000°K. Comparisons are made with available experimental data or calculated values and the general agreement is found generally good.

- 942 (COO--4116-2) INVESTIGATION OF ACCURACY AND COMPUTATION TIME OF A HIERARCHY OF GROWTH RATE DEFINITIONS. Maudlin, P.J.; Borg, R.C.; Ott, K.O. (Purdue Univ., Lafayette, IN (USA). School of Nuclear Engineering). Jul 1977. Contract EY-76-S-02-4116. 21p. Dep. NTIS, PC A02/MF A01.

A numerical illustration of the hierarchy of four logically different procedures for the calculation of the asymptotic growth of fast breeder fuel is presented. Each hierarchy level is analyzed in terms of accuracy and computational effort. Using the first procedure as reference, the fourth procedure, which incorporates the isotopic breeding worths, w vector*, requires a minimum amount of effort with a negligible decrease in accuracy.

- 943 (COO--4116-3) INVESTIGATION OF THE GROWTH RATE FOR JOINT FAST BREEDER REACTOR AND LIGHT WATER REACTOR OPERATION. Hanan, N.A.; Borg, C.R.; Ott, K.O. (Purdue Univ., Lafayette, IN (USA). School of Nuclear Engineering). Jul 1977. Contract EY-76-S-02-4116. 24p. Dep. NTIS, PC A02/MF A01.

An investigation of fuel consumption and breeding characteristics of FBR-LWR joint operation is presented. The FBR operates in a closed cycle with joint-reprocessing of core and blanket material. The LWR-portion that runs on FBR plutonium operates in an open cycle. The growth rate of the system is defined based upon the fact that the discharge from the system will make up a fraction of an identical system;

the system growth rate is found to have an almost linear dependence on the fraction of the LWR fed by plutonium from the FBR. The LWR growth rate, which is negative, is a constant and represents the fraction of the fuel burnt in the LWR=fraction that runs on FBR plutonium per year.

- 944 (COO--4171-3) HEAT TRANSFER CHARACTERISTICS OF A NON-BOILING POOL WITH SPATIALLY UNIFORM GAS INJECTION. Ganguli, A.; Luk, A.; Bankoff, S.G. (Northwestern Univ., Evanston, IL (USA). Dept. of Chemical Engineering). Dec 1978. Contract EN-77-S-02-4171. 36p. Dep. NTIS, PC A03/MF A01.

It is possible to encounter molten fuel pools in the course of events during a hypothetical Liquid Metal Fast Breeder Reactor accident. In such cases it may be important to estimate correctly the rate of melting of the pool walls and bottom, which are governed by the rate of heat transfer to those materials. The heat transfer characteristics of internally heated two-phase pools are thus of interest. Heat transfer measurements were made in the horizontal and downward directions both in transient and in steady state. The transient study involved slow cooling or heating of the pool liquid, while in the steady state case heat was provided by electrical immersion heaters. A permanent gas was injected from a distributed hypodermic tubing network with two hole sizes. Void fraction measurements were also made by static pressure probes. The data is reported as plots of Nusselts number versus Reynolds number and surface evaporation and entrainment effects for a boiling pool were estimated for comparison purposes.

- 945 (COO--4197-T1) SODIUM ION LEAK DETECTOR. QUARTERLY TECHNICAL PROGRESS REPORT, JANUARY 1-APRIL 1, 1978. Hrizo, J.; Bauerle, J.E.; Witkowski, R.E. (Westinghouse Research and Development Center, Pittsburgh, PA (USA)). 5 May 1978. Contract EN-77-C-02-4197. 9p. AT.

This program involves a basic technology development effort for the Westinghouse Sodium Ionization Detector (SID). The study consists of four task areas that will be performed over a two-year period. This effort will establish an in-situ operational testing/calibration method for the SID. It will test the performance/reliability characteristics of aged filaments and evaluate potential chemical interference effects. Quarterly and annual reports will be submitted detailing the work accomplished during the current period, problems encountered, and their resolutions.

- 946 (COO--4197-T2) SODIUM ION LEAK DETECTOR. QUARTERLY TECHNICAL PROGRESS REPORT, APRIL 1-JULY 1, 1978. Hrizo, J.; Bauerle, J.E.; Witkowski, R.E. (Westinghouse Research and Development Center, Pittsburgh, PA (USA)). 20 Jul 1978. Contract EN-77-C-02-4197. 5p. AT.

This program involves a basic technology development effort for the Westinghouse Sodium Ionization Detector (SID). The study consists of four task areas that will be performed over a two-year period. This effort will establish an in-situ operational testing/calibration method for the SID. It will test the performance/reliability characteristics of aged filaments and evaluate potential chemical interference effects. Quarterly and annual reports will be submitted detailing the work accomplished during the current period, problems encountered, and their resolutions.

heat transfer rates.

- 947 (COO--4197-T3) SODIUM ION LEAK DETECTOR. FIRST ANNUAL TECHNICAL PROGRESS REPORT, JANUARY 1-DECEMBER 31, 1977. Hrizo, J.; Bauerle, J.E.; Witkowski, R.E. (Westinghouse Research and Development Center, Pittsburgh, PA (USA)). 4 Jan 1978. Contract EN-77-C-02-4197. 12p. AT.

This program involves a basic technology development effort for the Westinghouse Sodium Ionization Detector (SID). The study consists of four task areas that will be performed over a two-year period. This effort will establish an in-situ operational testing/calibration method for the SID. It will test the performance/reliability characteristics of aged filaments and evaluate potential chemical interference effects. Quarterly and annual reports will be submitted detailing the work accomplished during the current period, problems encountered, and their resolutions.

- 948 (COO--4261-4) GROWTH OF MOLTEN CORE DEBRIS POOLS IN CONCRETE. PROGRESS REPORT, MARCH 1, 1977--NOVEMBER 30, 1977. Abdel-Khalik, S.I. (Wisconsin Univ., Madison (USA)). Dept. of Nuclear Engineering). Nov 1977. Contract EN-77-S-02-4261. 24p. Dep. NTIS, PC A02/MF A01.

Experiments have been conducted using a volumetrically-heated pool with noncondensable gas injection at the boundaries to simulate the heat transfer processes taking place in molten core debris/concrete systems. Measurements of the upward, downward, and sideward heat transfer rates have been made over wide ranges of internal Rayleigh number ($6.0 \times 10^5 < Ra < 2.7 \times 10^9$), superficial gas velocity ($0 < V_{sub} g / < 2.5 \text{ cm/min}$), and pool aspect ratio ($1.32 < W/L < 5.16$). Pools with either an isothermal solid upper boundary or a free surface have been examined. Detailed measurements of the temperature distribution within the pools have also been made. The results indicate that the downward heat transfer rates are significantly increased while the sideward heat transfer rates are slightly affected by gas injection. The heat transfer rates at the pool boundaries are sensitive to the upper surface boundary condition. Efforts are continuing on the second part of the present study where the phase-change and gas evolution processes are combined.

- 949 (COO--4261-8) CONVECTIVE HEAT TRANSFER CORRELATIONS FOR MOLTEN CORE DEBRIS POOLS GROWING IN CONCRETE. Felde, D.K.; Kim, H.S.; Abdel-Khalik, S.I. (Wisconsin Univ., Madison (USA)). Dept. of Nuclear Engineering). [nd]. Contract EN-77-S-02-4261. 5p. Dep. NTIS, PC A02/MF A01.

The heat and mass transfer processes taking place in volumetrically-heated pools are of interest because of their relevance in post accident heat removal analyses for fast reactors. Earlier studies on volumetrically-heated pools have focused on natural convection phenomena. These, however, do not apply for molten core debris pools growing in concrete since large amounts of gas will be released by the concrete. These gases will vigorously mix the pool. Knowledge of the heat transfer rates at the pool boundaries is necessary in order to assess the passive core retention capability of the concrete support structure below the reactor vessel. Hence, the main objectives of this investigation have been to experimentally simulate the gas evolution process at the pool boundaries and determine the effect of gas release on the heat transfer rates along the boundaries. Nusselt-type correlations have been obtained for the downward, sideward, and upward

- 950 (COO--4261-9) EFFECT OF OVERLAYING STEEL LAYERS ON THE GROWTH OF MOLTEN CORE DEBRIS POOLS IN CONCRETE. Paik, I.K.; Felde, D.K.; Abdel-Khalik, S.I. (Wisconsin Univ., Madison (USA)). Dept. of Nuclear Engineering). [nd]. Contract EN-77-S-02-4261. 5p. Dep. NTIS, PC A02/MF A01.

Following a hypothetical core disruptive accident (HCDA) in a fast reactor, the core debris may be in the form of a molten pool which is volumetrically heated by the radioactive decay of the dissolved fission products. The debris pool may be retained in the concrete support structure below the reactor vessel. It grows continually by melting the concrete which readily dissolves in the pool. The purpose of this research is to simulate the heat and mass transfer processes taking place in the molten steel/core debris/concrete system. The effects of the initially-lighter steel layer atop the pool and the subsequent inversion process on the transient radial and downward growth rates of the pool are investigated.

- 951 (COO--4261-10) RETENTION OF MOLTEN CORE DEBRIS POOLS IN COMPOSITE SACRIFICIAL BEDS. Paik, I.K.; Abdel-Khalik, S.I. (Wisconsin Univ., Madison (USA)). Dept. of Nuclear Engineering). [nd]. Contract EN-77-S-02-4261. 5p. Dep. NTIS, PC A02/MF A01.

Passive core catchers have been proposed as a means for retaining the core debris following a hypothetical core disruptive accident in a fast reactor. The core debris which may be in the form of a molten pool is passively retained in a sacrificial bed of high melting temperature material placed inside or outside the reactor vessel. Recently, it has been suggested that composite sacrificial beds with a low-melting-temperature zone inside the main high-melting-temperature bed may be more effective in retaining the pool than simple beds. It was argued that the inner zone would quickly melt and dilute the pool so that further growth into the main bed would be slowed. Hence, the main objective of this work has been to examine the composite bed concept to ascertain its efficacy vis-a-vis simple sacrificial beds.

- 952 (COO--4261-11) SIMULATION OF HEAT AND MASS TRANSFER PROCESSES IN MOLTEN CORE DEBRIS-CONCRETE SYSTEMS. Felde, D.K. (Wisconsin Univ., Madison (USA)). 1979. Contract EN-77-S-02-4261. 116p. Dep. NTIS, PC A06/MF A01.

Thesis.

The heat and mass transport phenomena taking place in volumetrically-heated fluids have become of interest in recent years due to their significance in assessments of fast reactor safety and post-accident heat removal (PAHR). Following a hypothetical core disruptive accident (HCDA), the core and reactor internals may melt down. The core debris melting through the reactor vessel and guard vessel may eventually contact the concrete of the reactor cell floor. The interaction of the core debris with the concrete as well as the melting of the debris pool into the concrete will significantly affect efforts to prevent breaching of the containment and the resultant release of radioactive effluents to the environment.

- 953 (COO--4261-12) GROWTH OF MOLTEN CORE DEBRIS POOLS IN CONCRETE. PROGRESS REPORT, APRIL 1--JUNE 30, 1979. Abdel-Khalik, S.I.

(Wisconsin Univ., Madison (USA). Dept. of Nuclear Engineering). 1 Jul 1979. Contract EN-77-S-02-4261. 48p. Dep. NTIS, PC A03/MF A01.

The heat and mass transfer processes taking place in molten core debris/concrete systems have been experimentally investigated. Two types of experiments have been conducted. The first experiment simulates the growth of a molten debris pool in a composite sacrificial bed. This experiment models debris pool growth in an inner, low-melting point, sacrificial material zone followed by a melting attack on the concrete bed. The purpose of the inner zone is to quickly melt and dilute the debris pool so that its subsequent downward growth in the concrete may be slowed. In the second experiment a two-layer immiscible liquid system is volumetrically heated and allowed to melt into a low-density gas-releasing solid bed which is miscible in the initially-higher-density bottom liquid. The solid melts, mixes with, and dilutes the bottom liquid pool until its density is lower than that of the top liquid. At this time pool inversion occurs and the immiscible liquid sinks to the bottom of the pool displacing the now lighter fuel-concrete simulant.

954 (COO--4261-14) GROWTH OF MOLTEN CORE DEBRIS POOLS IN CONCRETE. PART II. A. POOL GROWTH IN COMPOSITE BEDS; B. EFFECT OF OVERLAYING STEEL LAYERS. FINAL REPORT, MARCH 1, 1978-SEPTEMBER 30, 1979. Abdel-Khalik, S.I. (Wisconsin Univ., Madison (USA). Dept. of Nuclear Engineering). 26 Dec 1979. Contract EN-77-S-02-4261. 47p. Dep. NTIS, PC A03/MF A01.

The heat and mass transfer processes taking place in molten core debris/concrete systems have been experimentally investigated. Two types of experiments have been conducted. The first experiment simulates the growth of a molten debris pool in a composite sacrificial bed. This experiment models debris pool growth in an inner, low-melting point, sacrificial material zone followed by a melting attack on the concrete bed. The purpose of the inner zone is to quickly melt and dilute the debris pool so that its subsequent downward growth in the concrete may be slowed. In the second experiment a two-layer immiscible liquid system is volumetrically heated and allowed to melt into a low-density gas releasing solid bed which is miscible in the initially-higher-density bottom liquid. The solid melts, mixes with, and dilutes the bottom liquid pool until its density is lower than that of the top liquid.

955 (COO--4932.M001-2) ANALYSIS OF FLOW IMPEDANCE PHENOMENA. FINAL REPORT, JUNE 1, 1978-JULY 31, 1979. Bishop, A.A. (Pittsburgh Univ., PA (USA)). Dec 1979. Contract ET-78-S-02-4932. 102p. Dep. NTIS, PC A06/MF A01.

Pressure drop data obtained during testing of the MARK I, II and IIA fuel assemblies in the Argonne National Laboratory liquid metal core component loop (CCTL) were analyzed. It is concluded that the previously unexplained pressure drop increase measured across the MARK II and MARK IIA assemblies was caused by metallic particulate fouling; the source was mechanical-chemical wear of the loops pump journal shaft. The implication to liquid metal breeder reactor (LMFBR) operation is that if a continuous source of metallic particles exists upstream of the core, a pressure drop increase can be expected. The size of the deposited particles should be larger than the thickness of laminar sublayer on a fuel rod. It is also

concluded that the most probable source of metallic particulates will come from component failure and not from normal corrosion products. A small amount of deposited particulates can produce a relatively high pressure drop increase.

956 (COO--4954-1(Vol.1)) ENERGY ECONOMIC DATA BASE (EEDB) PROGRAM: PHASE I, VOLUME I. FINAL REPORT. (United Engineers and Constructors, Inc., Philadelphia, PA (USA)). Dec 1979. Contract EN-78-C02-4954. 306p. (UE/C-DOE--790930(Vol.1)). Dep. NTIS, PC A14/MF A01.

The Energy Economic Data Base Program, which deals with the development of cost data for nuclear and comparison electric power generating stations, provides periodic updates of technical and cost (capital, fuel, and operating and maintenance) information of significance to DOE. The information allows for evaluation and monitoring of US civilian nuclear power programs and provides a consistent means of evaluation the nuclear option against alternatives. Currently, the EEDB contains 6 nuclear electrical generating plant technical models and 5 comparison coal-fired electrical generating plant technical models. Each of these technical plant models is a complete conceptual design for a single unit, steam electric power generating station located on a standard, hypothetical Middletown site. A description of the site is provided in Appendix A-1 (Volume 2) for nuclear plants, and Appendix A-2 (Volume 2) for coal-fired plants. The EEDB also includes a conceptual design of a coal liquefaction plant for comparison purposes. Volume 1 provides a description of the current Data Base, as of September 30, 1978: gives assumptions and ground rules for the initial-cost update; summarizes the initial cost update, with cost results tabulated; details the initial update of the technical conceptual design, the capital cost, the quantities of commodities and their unit costs, and craft labor man hours and costs for each EEDB program model; and details the fuel-cycle-cost initial update and the operating- and maintenance-cost initial update. Finally, an extensive list of references and a glossary are presented.

957 (COO--4954-1(Vol.2)) ENERGY ECONOMIC DATA BASE (EEDB) PROGRAM: PHASE I, VOLUME II. FINAL REPORT. (United Engineers and Constructors, Inc., Philadelphia, PA (USA)). Dec 1979. Contract EN-78-C-02-4954. 386p. (UE/C-DOE--790930(Vol.2)). Dep. NTIS, PC A14/MF A01.

This Volume II of Phase I of the Energy Economic Data Base Program contains appendices. Appendix A-1 provides the site and environmental data, derived from Appendix A of Guide for Economic Evaluation of Nuclear Reactor Plant Designs, USAEC Report NUS-531, modified to reflect current requirements. These data form the bases of the criteria used for designing the facility and for evaluating the routine and accidental release of radioactive liquids and gases to the environment. Appendix A-2 provides the site and environmental data as derived from Appendix A of NUS-531, and modified to reflect coal-plant siting, forming the bases of the criteria used for designing the facility and for evaluating the release of liquids and gases to the environment. A description of the topography of the hypothetical city, Middletown, is given. Appendix B provides an overall summary of the conclusions of NUS' work on all NUS tasks in support of the nuclear fuel-cycle work in Phase I. Appendix C-1 introduces the concepts involved and addresses methods of calculation

of fixed charges applicable to investor-owned utilities, as used in the EEDB. Appendix C-2 consists of review and revision of each plant's fuel cycle and operating and maintenance costs in accordance with the EEDB update procedures. In Appendix D, NSSS Capital Costs for a Mature LMFBR Industry, much information is provided on plant description, cost estimate, comparison and discussion, drawings, and equipment list. (MCW)

- 958 (COO/4116--1) IMPACT OF DELAYS IN PLUTONIUM USE ON THE STATIONARY GROWTH RATE OF FAST BREEDER REACTOR FUEL. Borg, R.C.; Ott, K.O. (Purdue Univ., Lafayette, IN (USA). School of Nuclear Engineering). Jul 1977. Contract EY-76-S-02-4116. 55p. Dep. NTIS, PC A04/MF A01.

The hierarchy of the four growth rate expressions originally derived from an instantaneous reuse scheme is expanded to account for finite burnup in the core and blanket, β -decay of ^{241}Pu , core and blanket loading schemes, reuse delays due to reprocessing and fabricating fuel and external fuel cycle losses. The most general growth rate expression, obtained from the asymptotic slope of the accumulating fuel material in an expanding park of breeder reactors, is formally the same in both cases. Formulation of the growth rate based on the condensation of the detailed information of the equilibrium fuel cycle for a single reactor, is more complicated than without delays due to the composition difference between the average residing and excess discharge material. The third growth rate expression results from a slightly more complicated fuel-cycle eigenvalue problem than without delays. The last definition employs isotopic breeding worth factors obtained from the adjoint fuel cycle eigenvalue problem.

- 959 (COO/4171--1) FORMATION AND STABILITY OF CRUST IN MOLTEN POOLS. TECHNICAL PROGRESS REPORT, FEBRUARY 1, 1977--OCTOBER 1, 1977. Bankoff, S.G.; Ganguli, A. (Northwestern Univ., Evanston, IL (USA). Dept. of Chemical Engineering). 1977. Contract EN-77-S-02-4171. 43p. Dep. NTIS, PC A03/MF A01.

The objective of the research is to study the formation and stability of a freezing-melting process under well-defined hydrodynamic and thermal boundary conditions simulating LMFBR HCDA phenomena. A hot liquid, such as molten UO_2 , comes into contact with a cold solid, such as stainless steel forming a solid crust at the interface, possibly accompanied by the simultaneous formation of a melt layer. The stability and rate of growth of the crust and underlying melt layer are thus of concern. A steady flowpast an initially flat slab of the cold solid is considered. The experimental apparatus consists of a horizontal rectangular test section in which a frozen slab of n-decane 18-in. long and 4-in. wide, supported on a refrigerated copper block, is exposed to a flowing stream of cold water. Temperature profiles in the decane will be measured, as well as the rate of liquid decane collection, and the ice crust and melt layer profiles, if stable. These will be compared with theoretical predictions, following an integral boundary layer method. Current efforts are focussed on improving the mold release properties and other debugging operations.

- 960 (CRBRP--1(Vol.1)) CRBRP SAFETY STUDY: AN ASSESSMENT OF ACCIDENT RISKS IN THE CRBRP. VOLUME I. MAIN REPORT. (Clinch River Breeder Reactor Plant Project Office, Oak Ridge, TN (USA)). Mar 1977. Contract EY-76-C-15-1200.

222p. Dep. NTIS, PC A10/MF A01.

The objective of the CRBRP risk assessment is to provide a realistic evaluation of the risk to the public from the Clinch River Breeder Reactor Plant, to place that risk in perspective relative to other societal risks, and to provide a basis for assessing the comparability of the risk from the CRBRP with risks associated with previously licensed reactors. The assessment formally and systematically evaluated the CRBRP risk and showed that (1) the risk associated with postulated accidents of the CRBRP is negligible when compared with non-nuclear risks to which the local population is already exposed, (2) CRBRP risks are comparable to those from light-water reactors (LWRs) of the current generation as characterized in the Reactor Safety Study (RSS), and (3) the results of this study and those presented in the RSS reveal that the societal risk from currently defined nuclear power systems (both LWRs and the CRBRP) is extremely small.

- 961 (CRBRP--1(Vol.2)) ASSESSMENT OF ACCIDENT RISKS IN THE CRBRP. VOLUME 2. APPENDICES. (Clinch River Breeder Reactor Plant Project Office, Oak Ridge, TN (USA)). Mar 1977. Contract EY-76-C-15-1200. 750p. Dep. NTIS, PC A19/MF A01.

Appendices to Volume I include core-related accident=sequence definition, CRBRP risk=assessment sequence=probability determinations, failure=probability data, accident scenario evaluation, radioactive material release analysis, ex-core accident analysis, safety philosophy and design features, calculation of reactor accident consequences, sensitivity study, and risk from fires.

- 962 (CRBRP-ARD--0119) CLINCH RIVER BREEDER REACTOR PLANT. PRELIMINARY ANALYSIS OF HEAT GENERATING BLOCKAGES IN CRBRP FUEL AND RADIAL BLANKET ASSEMBLIES TO DETERMINE DETECTION REQUIREMENTS. Brown, R.G. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Dec 1977. Contract EY-76-C-15-0003; EY-76-C-15-2395. 64p. AT.

Heat generating blockages in CRBR fuel and radial blanket assemblies during steady state operation have been analyzed. Blockages of both a single subchannel and six subchannels completely encircling one pin have been investigated. The blockage porosity and particle size have been treated parametrically. The results itemized in Tables IV through IX have been provided to assist in determining the sensitivity requirements of the delayed neutron monitoring system. For purposes of blockage detectability by the delayed neutron monitors, the total surface area of all particles within the blockage which are exposed to the sodium is the important parameter. Conclusions of this study are discussed.

- 963 (CRBRP-ARD--0132) CLINCH RIVER BREEDER REACTOR PLANT. A PRELIMINARY EVALUATION OF THE CRBRP NATURAL CIRCULATION DECAY HEAT REMOVAL CAPABILITY. Lowrie, R.R.; Severson, W.J. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Nov 1977. Contract EY-76-C-15-2395; EY-76-C-15-0003. 291p. AT.

Using the "LMFBR Demonstration Plant Simulation Model (DEMO)" a preliminary evaluation of the Clinch River Breeder Reactor Plant decay heat removal capability by natural circulation has been made. This assessment was made in response to the design requirement that, although the plant is a forced circulation plant, emergency cooling shall be

based on natural circulation in all main heat transport loops. Based on this evaluation, the preliminary conclusions are: (1) The arrangement of components within the plant (i.e., the relative elevations of the IHX with respect to the core, the steam generator modules with respect to the IHX and the steam drum with respect to the evaporator modules) will provide thermal driving heads necessary to promote adequate flows in the natural circulation mode. (2) The specified pump coastdown characteristics satisfy the minimum flow decay requirements immediately following a plant trip. (3) While the CRBRP is a forced circulation plant, its heat transport and steam generator systems will provide a desirable margin in decay heat removal by their natural circulation capability.

- 964 (CRBRP-ARD--0147) CLINCH RIVER BREEDER REACTOR PLANT. INTERNAL/EXTERNAL CLADDING DEGRADATION. Travis, M.L. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1977. Contract EY-76-C-15-1200;EY-76-C-15-2395. 63p. AT.

Cladding degradation of CRBRP fuel cans due to sodium-cladding interactions, fuel-cladding interactions, and fretting wear is discussed.

- 965 (CRBRP-ARD--0149) CLINCH RIVER BREEDER REACTOR PLANT. WIRE WRAP CLADDING INTERACTION IN LMFBR FUEL RODS. Schwegler, E.C. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Dec 1977. Contract EY-76-C-15-2395;EY-76-C-15-0003. 107p. AT.

Most of the currently accepted designs of the fuel and blankets for the LMFBR utilize helically wound wires for spacing the individual rods within a rod bundle. The fuel assembly design for the CRBRP also utilizes a wire wrap as a rod spacer/support system. The purposes of this report are to delineate the functional objectives of the wire-wrap design, to describe the methods applied for evaluating the performance adequacy of the wire wrap system for the CRBRP fuel assembly design, and to present current experimental results concerning wire wrap performance.

- 966 (CRBRP-ARD--0168) CLINCH RIVER BREEDER REACTOR PLANT. IMPACT OF FUEL DENSIFICATION ON CRBRP FUEL PERFORMANCE. Bishop, B.A. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jun 1977. Contract EY-76-C-14-2395. 63p. TIC.

A review of both LWR and LMFBR mixed-oxide fuel literature shows that fuel densification mechanisms are similar. The primary factor is unirradiated porosity and the associated fuel microstructure. If fuel densification were to become a problem in CRBRP, there is substantial evidence in the literature that densification could be controlled by changes in the fuel manufacturing process. Based upon low burnup FFTF vendor fuel data, which is CRBRP prototypic, and high burnup nonprototypic data, which overpredicts CRBRP densification, from the EBR-II Reference Fuels Irradiation Program, a conservative design bases for CRBRP fuel densification was derived. The maximum CRBRP fuel pellet length decrease was set at 0.4 to 1.1% at 15 to 21 MWd/Kg local fuel burnup. These values were shown to envelop or exceed the values in the published literature. The maximum column shortening in the peak power fuel rod was 0.352 inches while the core average value is 0.220 inches.

- 967 (CRBRP-ARD--0173) CLINCH RIVER BREEDER

REACTOR PLANT. PLAN FOR INSERVICE INSPECTION. Blinn, H.O.; Chase, W.L. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Sep 1977. Contract EY-76-C-15-2395;EY-76-C-15-0003. 45p. AT.

The plan describes the approach to be followed in specifying the type, extent, and frequency of inservice inspections to be performed on the Clinch River Breeder Reactor Plant (CRBRP). The plan also gives the basis for the approach and specifies requirements for inspection. In addition, a preliminary listing of examinations to be performed on components and systems which make up the coolant boundary is included. The inservice inspection requirements given are based on protection of public health and safety.

- 968 (CRBRP-ARD--0175) CLINCH RIVER BREEDER REACTOR PLANT. MAXPRES-2: COMPUTER CODE VERIFICATION. Ranatza, S. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jul 1977. Contract EY-76-C-15-2395;EY-76-C-15-1200;EY-76-C-15-0003. 31p. AT.

During an energetic LMFBR Hypothetical Core Disruptive Accident (HCDA), the rapid expansion of the HCDA bubble would accelerate the slug of sodium above the core. As the sodium rises, the argon cover gas would be compressed. After the sodium slug impacts the reactor vessel closure head, the sodium would penetrate the spaces within the head structure and the head-mounted component annuli. The argon cover gas in the annuli would be further compressed by the penetrating sodium. The MAXPRES-2 code analyzes the argon and sodium flow in the annuli to determine maximum annuli pressures and, ultimately, the capability of various seals and annuli geometries for mitigation of HCDA consequences. During the development of the code, various comparisons and checks were made to verify that the code programming and formulation accurately represented the mathematical model. Two of these comparisons and checks which were made to the key parts of the code are described in detail. A comparison was made between a separately programmed simplified computer calculation (MPCK) and MAXPRES-2 to check the integration subroutine and the momentum equation subroutine. A separately programmed simplified calculation (AFLOCK) of gas flow equations was made to check the MAXPRES-2 AFLO subroutine.

- 969 (CRBRP-ARD--0183) CLINCH RIVER BREEDER REACTOR PLANT. CRBRP PIPING INTEGRITY: AN EVALUATION OF POSTULATED SUPPORT FAILURES. Mallett, R.H.; Nair, B.R.; Mello, R.M. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Sep 1977. Contract EY-76-C-15-1200;EY-76-C-15-2395. 78p. AT.

The CRBRP Project position on piping integrity is that the probability of a gross failure of the heat transport system piping within containment is low enough to preclude such an event from being considered a design basis event. As part of the overall effort to establish this position, a special evaluation of postulated support failures in the primary heat transport system piping was performed even though such failures are not included in the Plant Duty Cycle. The results of the preliminary evaluation, which was focused on the primary cold leg between the intermediate heat exchanger and the reactor vessel inlet, are presented in the report. It is concluded that the integrity of the primary cold leg piping will be preserved even in the unlikely event of the failure of a snubber or pipe hanger.

970 (CRBRP-ARD--0184) CLINCH RIVER BREEDER REACTOR PLANT. NUMERICAL EVALUATION OF AN ELBOW ELEMENT FOR INELASTIC FLEXIBILITY ANALYSIS OF PIPING. Scheller, J.D.; Mallett, R.H.; Nair, B.R. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Sep 1977. Contract EY-76-C-15-2395. 109p. AT.

The PIRAX computer program embodies a simplified model for predicting inelastic behavior of piping elbows at elevated temperature. This model is evaluated for use in inelastic flexibility analysis of LMFBR piping by comparing results obtained using PIRAX with results obtained from more complex analytical models and tests. Isolated piping elbows are examined under in-plane and out-of-plane loading. Plasticity, thermal creep under load-controlled loading and relaxation by thermal creep under deformation-controlled loading are considered. The load versus deformation behavior obtained using the simplified model for piping elbows is judged to be a useable approximation. Based upon this evaluation, it is concluded that the inelastic flexibility analysis capability of the PIRAX computer program should be useful for detecting elastic followup in LMFBR piping systems and for classifying forces induced in LMFBR piping systems into load-controlled and deformation-controlled categories.

971 (CRBRP-ARD--0185(Vol.1)) CLINCH RIVER BREEDER REACTOR PLANT. INTEGRITY OF PRIMARY AND INTERMEDIATE HEAT TRANSPORT SYSTEM PIPING IN CONTAINMENT. Mallett, R.H.; Nair, B.R. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1977. Contract EY-76-C-15-1200;EY-76-C-15-2395. 330p. AT.

A comprehensive and detailed assessment of the primary and intermediate Heat Transport System (HTS) piping in containment has led to the conclusion that the integrity of this HTS piping is assured such that there is a negligible likelihood of a large break. This conclusion is based upon the following: (1) The overall system design incorporates many features which prevent the occurrence, and limit the consequences, of leaks in this HTS piping; (2) The highest quality engineering standards are applied by the HTS Piping Design Specification which provides a complete basis for construction including the materials, design, fabrication, examination, testing, inspection and certification; (3) A comprehensive quality assurance program assures that the specified engineering standards are met throughout the construction, installation, operation and maintenance of this HTS piping; (4) A compilation of operating experience with liquid metal piping governed by various quality standards has shown that, although fatigue crack leakage incidents have occurred, no large breaks have occurred in liquid metal piping; (5) A detailed structural design analysis has shown that the piping will comply with all of the Design Specifications and ASME Code requirements for Class 1 piping; (6) A detailed fracture mechanics evaluation has shown that significant fatigue crack propagation will not occur during the entire plant lifetime even if an initial defect much larger than quality standards allow is postulated; and (7) A comprehensive technology program has shown that even if a defect did grow significantly, it would penetrate the pipe wall and result in detectable leakage without risk of a large break in this HTS piping.

972 (CRBRP-ARD--0185(Vol.2)) CLINCH RIVER BREEDER REACTOR PLANT. INTEGRITY OF PRIMARY AND INTERMEDIATE HEAT TRANSPORT SYSTEM PIPING IN CONTAINMENT. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1977. Contract EY-76-C-15-1200;EY-76-C-15-2395. 180p. AT.

Appendix titles include the development program for ultrasonic examination of primary piping, the development program for review of sodium piping experience, the development program for long term thermal aging effects on welds, the development program for sodium leak detection verification, aerosol and radiation monitor sensitivity and response time calculations, an evaluation of postulated support failures, and sodium effects on long term material properties.

973 (CRBRP-ARD--0186) CLINCH RIVER BREEDER REACTOR PLANT. PRELIMINARY TESTING OF THE DIRECT HEAT REMOVAL SERVICE IN A 1/21 SCALE MODEL. Marinkovich, P.S. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1977. Contract EY-76-C-15-1200;EY-76-C-15-2395. 31p. AT.

A Direct Heat Removal Service (DHRS) test was performed in the ARD 1/21 scale Outlet Plenum Feature Model (OPFM) to determine the amount of cold make-up flow which will flow directly from the make-up nozzle to the DHRS overflow nozzle. This flow is termed short circuit flow and it is assumed that it does not contribute to cooling of the primary heat transport coolant. The results show that five to six percent of the make-up flow will short circuit at prototypic DHRS operating conditions; this is concluded to be a conservative confirmation of the design since an allowance of twenty percent has been used in sizing the DHRS.

974 (CRBRP-ARD--0188) CLINCH RIVER BREEDER REACTOR PLANT. EFFECTS OF THERMAL AGING ON CRBRP PRIMARY PIPING. Leasure, R.A. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1977. Contract EY-76-C-15-1200;EY-76-C-15-2395. 60p. AT.

The primary piping systems of Liquid Metal Fast Breeder Reactors (LMFBR's) are designed for many years of safe and maintenance free, high temperature operation. The materials being used to construct these systems are primarily the austenitic stainless steels which are selected for their good high temperature mechanical properties and compatibility with the liquid sodium environment. For the Clinch River Breeder Reactor Plant (CRBRP), which is a demonstration plant in the LMFBR program with a thirty year design life, the proposed primary piping materials include Type 304 and Type 316 stainless steels. As in any system that has to operate safely and without maintenance for many years, it is necessary to examine the effects of long term service on the properties of the materials in the CRBRP piping. This report examines the effects of the major factor contributing to changes in material properties of the primary piping over the thirty year service life, i.e., thermal aging. Included in this examination are the effects of thermal aging on microstructure, tensile properties, creep, fatigue and the fracture toughness of both base metal and weldments.

975 (CRBRP-ARD--0192) CLINCH RIVER BREEDER REACTOR PLANT. CRACK GROWTH CALCULATIONS FOR PRIMARY PIPING OF CRBRP. Wilson, W.K. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Dec 1977.

Contract EY-76-C-15-2395;EY-76-C-15-0003.
35p. AT.

The growth patterns under cyclic loading of surface cracks which are postulated to exist in the primary piping of the Clinch River Breeder Reactor Plant (CRBRP) are investigated. An analytical model which can be used to calculate the crack growth history of surface cracks is presented. The model is based upon the fracture mechanics approach for calculating crack growth rates. The applicability of the model is checked in part by comparing calculated data with experimentally measured data. Using the model, crack growth calculations are made under the expected duty cycles for the primary piping of the CRBRP. Special emphasis is placed on calculating the crack length at the time of wall penetration.

976 (CRBRP-ARD--0211) CLINCH RIVER BREEDER REACTOR PLANT PROJECT. 1977 TECHNICAL PROGRESS REPORT. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1977. Contract EY-76-C-15-2395;EY-76-C-15-0003. 368p. AT.

In this report period, much of the basic design and documentation were completed and refined. Analysis and testing methods were improved and applied to many components. Much design and analysis effort was applied to the alternate fuel management scheme, which is an arrangement of fuel and blanket assemblies that could improve the breeding ratio, reduce peak neutron flux levels, and reduce the number of fuel and control assemblies required as compared with the original fuel management scheme. Fabrication of many components is well along. The reactor vessel, for example, was more than half completed by the end of the report period. Fabrication and checkout were completed for the first prototype of the secondary control rod system, and testing was initiated.

977 (CRBRP-ARD--0230) CLINCH RIVER BREEDER REACTOR PLANT PROJECT. 1978 TECHNICAL PROGRESS REPORT. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1978. Contract EW-76-C-15-2395;EW-76-C-15-0003. 379p. AT.

This technical progress report on the Clinch River Breeder Reactor Plant Project describes the objectives, design decisions, and major accomplishments achieved in the planning, organizing, design, and licensing of the Clinch River Breeder Reactor Plant (CRBRP). In this report period, fabrication of components continued. Several components were completed, the most significant being the cold-leg check valves. Fabrication of the reactor vessel passed the 80-percent completion stage. All licensing action remains suspended until decisions regarding the Project have been made by the Administration and Congress.

978 (CRBRP-GEFR--00025) ANALYSIS OF THE ZPPR-7 CRITICAL EXPERIMENTS. (Clinch River Breeder Reactor Plant Project Office, Oak Ridge, TN (USA)). Jul 1977. Contract EY-76-C-15-2395. 99p. AT.

Calculations of the ZPPR-7 critical experiments have provided estimates of design bias factors and uncertainties for standard nuclear design tools used in the analysis of radial heterogeneous cores. The analysis concentrated on calculations of criticality, neutron reaction rate distributions, sodium void worths, small sample reactivity worths, and control and refueling worths. Differences were observed in the accuracy of the predicted nuclear performance of heterogeneous cores

relative to conventional designs. However, nothing was found which would impact the viability of using standard design techniques for radial heterogeneous designs since the calculated differences were generally found to be in a conservative direction.

979 (CRBRP-GEFR--00100) CLINCH RIVER BREEDER REACTOR PLANT. DNB EFFECTS TEST PROGRAM. FINAL REPORT. Stephens, P.B.; Rodgers, D.N.; Valente, C.G.; Hampton, L.V.; Wolf, S.; Holmes, D.H.; Rush, D.J. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jun 1977. Contract EY-76-C-15-2395. 110p. AT.

The CRBR Plant evaporators will operate with departure from nucleate boiling (DNB) in the tubes over the power range of approximately 60 to 100%. When the 2:1 recirculation ratio was selected for the plant evaporators in 1974, potential damage mechanisms associated with DNB were identified as increased local corrosion in the DNB zone due to thermal strain cycling-induced exfoliation of the protective oxide layer and/or attack by soluble corrodents concentrated in the water. Thus, the DNB Effects Test Program was conducted to verify that excessive damage to the evaporator tubes would not be produced by operation with DNB or liquid film dryout. A sodium-heated single-tube test section with a 16-foot (4.9 m) active heat transfer length was designed, fabricated, and operated under test conditions to simulate "worst case" thermal and water chemistry conditions expected for the plant evaporators. Operation under these endurance conditions established the nucleate boiling and DNB regions, and the initial part of the post-DNB boiling region of the plant evaporator within the test section. The test section was operated for a total of 4181 hours of steaming test time, of which 2820 hours of steaming were with DNB held within a 24-inch (61 cm) test zone. The principal findings after post-test examination were: (1) no localized damage or accelerated corrosion, (2) long-term life, in the order of 30 years, is predicted based on extrapolation of the test data, and (3) deposition (fouling) on the tube wall was minimal.

980 (CRBRP-GEFR--00103) ANALYSIS OF HYPOTHETICAL CORE DISRUPTIVE EVENTS IN THE CLINCH RIVER BREEDER REACTOR PLANT. McElroy, J.L.; Switick, D.M.; Brown, N.W.; Chou, F.M.; Damerow, W.P.; Rodgers, F.C. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Apr 1978. Contract EY-76-C-15-2395. 551p. AT.

The results of hypothetical core disruptive event analyses for the Clinch River Breeder Reactor Plant performed during the period of January 1975 to January 1977 (with the SAS3A and VENUS II computer codes) are reported. The analytical results cover a large number of parametric cases including variations in design parameters, initiating conditions and phenomenological assumptions. Reactor core configurations at the beginning and end of an equilibrium cycle are evaluated. Accident progression branch points leading to either hydraulic pressure (within fuel assembly duct structural capacity) or hydrodynamic pressure (exceeding fuel assembly duct structural capacity) fuel disruption modes are evaluated. The phenomenological processes of molten fuel pools are described and the consequences of energetic recriticality, including Behrens' effects, are examined. The energetic consequences are evaluated based upon both fuel expansion thermodynamic work potential and a relative probability assignment. It is

concluded that the structural loads which result from 101 megajoules of available expansion work at sodium slug impact on the reactor closure head (equivalent to 661 megajoules of fuel expansion work to one atmosphere) is an adequate energetic consequence envelope for use in specifying the Structural Margins Beyond the Design Basis.

- 981 (CRBRP-GEFR--00323) ANALYSIS OF ZPPR-6 CRITICAL EXPERIMENTS. Lee, B.W.; Stewart, S.L. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jan 1978. Contract EY-76-C-15-2395. 143p. AT.

This report presents analysis of critical experiments performed in the Zero Power Plutonium Reactor (ZPPR) in support of the Clinch River Breeder Reactor Project (CRBRP) initial core critical, power distribution, and neutron and gamma flux distribution. The major emphasis is on analysis of the CRBRP Engineering Mockup Critical (EMC) Experiment Phase 2A also identified as ZPPR Assembly 6.

- 982 (CRBRP-GEFR--06465) HISTORY OF THE WATER CHEMISTRY FOR THE FEW TUBE TEST MODEL. Moss, S.A.; Simpson, J.L. (General Electric Co., San Jose, CA (USA). Advanced Reactor Systems Dept.). Sep 1979. Contract EY-76-C-03-0189-061. 42p. Dep. NTIS, PC A03/MF A01.

The water chemistry activities carried out in support of the Few Tube Test are described. This test was conducted to provide design confirmation data for the Clinch River Breeder Reactor Project (CRBRP) steam generators. Proposed CRBRP chemistry was followed; all volatile treatment (AVT) of water was carried out with on-line monitoring capability.

- 983 (CRBRP-GEFR-SP--036) EVALUATION OF STRAIN CYCLING EFFECTS IN THE DNB ZONE OF A SODIUM-HEATED STEAM GENERATOR. Roberts, J.M.; Chu, C.L.; Wolf, S.; Magee, P.M. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Nov 1977. Contract EY-76-C-0893-027. 23p. (CONF-780203--8). TIC \$4.00.

From 1. joint U.S./Japan LMFBR steam generator seminar; Tokyo, Japan (13 Feb 1978).

Two sources of strain cycling in LMFBR steam generator tubing are identified as resulting from departure from nucleate boiling (DNB); namely, alternate transposition of rivulets and dry areas in the DNB zone itself and gross movement of the entire DNB zone due to system parameters disturbances originating in the controllers for the system. Experimental evidence and observations are used to support the analytical development of a finite element rivulet model which is used to calculate tube fatigue life. The gross movement is analytically represented by a finite element type trapezoidal wave front model. Results of the study show that lifetime operation (30 years) of the plant at the worst case plant power level (64%) for tube fatigue is safe. This assessment is shown to be conservative both from the standpoint of analysis assumptions and plant operations.

- 984 (CRBRP-GEFR-SP--040) DEVELOPMENT OF LEAK DETECTION SYSTEMS FOR LMFBR STEAM GENERATORS IN THE U.S. Magee, P.M.; Gerrels, E.E.; Greene, D.A. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Nov 1977. Contract EY-76-C-03-0893-027. 23p. (CONF-780203--7). AT.

From 1. joint U.S./Japan LMFBR steam generator seminar; Tokyo, Japan (13 Feb

1978).

Leak detection for Liquid Metal Fast Breeder Reactor steam generators is an important economic factor in the shutdown, repair and restart of a plant. The paper summarizes U.S. progress in three key technology areas involved in leak detection development: (1) chemical (H_2 and O_2) leak detection meters; (2) acoustic leak detection/location techniques, and (3) data management for plant operations. The status, design aspects, and applications of leak detector technology for LMFBR plants are discussed.

- 985 (CRBRP-GEFR-SP--139) INTERNAL FLUID FLOW MANAGEMENT ANALYSIS FOR CLINCH RIVER BREEDER REACTOR PLANT SODIUM PUMPS. Cho, S.M.; Zury, H.L.; Cook, M.E.; Fair, C.E. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Dec 1978. Contract EY-76-C-03-0893-001. 27p. (CONF-781202--4). Dep. NTIS, PC A03/MF A03.

From ASME meeting; San Francisco, CA, USA (10 Dec 1978).

The Clinch River Breeder Reactor Plant (CRBRP) sodium pumps are currently being designed and the prototype unit is being fabricated. In the design of these large-scale pumps for elevated temperature Liquid Metal Fast Breeder Reactor (LMFBR) service, one major design consideration is the response of the critical parts to severe thermal transients. A detailed internal fluid flow distribution analysis has been performed using a computer code HAFMAT, which solves a network of fluid flow paths. The results of the analytical approach are then compared to the test data obtained on a half-scale pump model which was tested in water. The details are presented of pump internal hydraulic analysis, and test and evaluation of the half-scale model test results.

- 986 (CRBRP-GEFR-SP--176) CLINCH RIVER BREEDER REACTOR PLANT STEAM GENERATOR: FEW TUBE TEST MODEL POST TEST EXAMINATION. Impellezzeri, J.R.; Camaret, T.L. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 13 Sep 1979. Contract EY-76-C-03-0189-061. 8p. (CONF-800401--10). NTIS, PC A02/MF A01.

From 2. international conference of liquid metal technology for energy systems; Richland, WA, USA (20 Apr 1980).

The Steam Generator Few Tube Test (FTT) is part of an extensive testing program being carried out in support of the Clinch River Breeder Reactor Plant (CRBRP) steam generator design. The testing of full-length seven-tube evaporator and three-tube superheater models of the CRBRP design was conducted to provide steady-state thermal/hydraulic performance data to full power per tube and to verify the absence of multi-year endurance problems. The problems encountered with the mechanical features of the FTT model design which led to premature test termination and the results of the post-test examination are described.

- 987 (CRBRP-GEFR-SP--00043) COMPARISON OF RESULTS FROM LARGE-LEAK SODIUM/WATER TESTS AT LLTR WITH ANALYSIS. Meyer, R.A.; Bradfute, J.O.; Lee, L.S.; Regimbal, J.J. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Feb 1978. Contract EY-76-C-03-0893. 27p. (CONF-780203--6). AT.

From 1. joint U.S./Japan LMFBR steam generator seminar; Tokyo, Japan (13 Feb 1978).

Pre-test analyses performed for the Series I Large Leak Test Rig (LLTR) tests using the

TRANSWRAP II computer code are described. These pre-test predictions are compared to the test data and significant differences between the pre-test analyses and the data are evaluated. These evaluations include three major areas: the source term, acoustic transmission of pressure disturbances, and the response of the LLTR sodium-water reaction relief system. The degree of conservatism in the analytical predictions is assessed and some possible areas for reducing conservatism in the analysis are identified.

- 988 (CRBRP-PMC--77-04) CLINCH RIVER BREEDER REACTOR PLANT. TECHNICAL REVIEW, FALL 1977. (Clinch River Breeder Reactor Plant Project Office, Oak Ridge, TN (USA)). 1 Oct 1977. Contract EY-76-C-15-1200. 47p. AT.

Included are sections dealing with seismic design of cantilever cylindrical shells, uniaxial compressive strengths of concrete for temperatures reaching 1033K, and practical aspects of probabilistic structural reliability analyses. (DG)

- 989 (CRBRP-PMC--78-01) CLINCH RIVER BREEDER REACTOR PLANT. TECHNICAL REVIEW, WINTER 1978. (Clinch River Breeder Reactor Plant Project Office, Oak Ridge, TN (USA)). 1 Jan 1978. Contract EY-76-C-15-1200. 66p. AT.

Information on the Clinch River Breeder Reactor is presented concerning accident analysis and safety studies; design and development of the ex-vessel transfer machine; and design and fabrication considerations for the ex-vessel storage tank.

- 990 (CRBRP-PMC--79-01) TECHNICAL REVIEW. (Project Management Corp., Oak Ridge, TN (USA)). Win 1979. Contract EY-76-C-15-1200. 55p. AT.

This quarter the Project continued the major tasks of plant design, component procurement and fabrication. As of December 30, 1978, the design was over 66% complete, research and development borne by the Project 79.7% complete and component fabrication about 37% complete. Information is presented concerning the need for the CRBRP Reactor and alternate core designs and fuels for the CRBRP Reactor.

- 991 (CRBRP-PS--01) OVERVIEW OF THE CRBRP SAFETY STUDY. Piper, H.B.; Conrad, L.L.; Buhl, A.R.; Wood, P.J.; Leaver, D.E.W. (Clinch River Breeder Reactor Plant Project Office, Oak Ridge, TN (USA); Energy Research and Development Administration, Oak Ridge, TN (USA); Oak Ridge Operations Office; Westinghouse Electric Corp., Madison, PA (USA); Advanced Reactors Div.; Science Applications, Inc., Palo Alto, CA (USA)). Jun 1977. Contract EY-76-C-15-2395. 31p. (CONF-77061--16). Dep. NTIS, PC A03/MF A01.

From American Nuclear Society annual meeting; New York, NY, USA (12 Jun 1977).

A Safety Study has been conducted for the Clinch River Breeder Reactor Plant (CRBRP). This paper presents a review and discussion of objectives, methods, techniques, and results of that study. The CRBRP Safety Study was conducted to: (1) provide a realistic assessment of accident risks associated with operation of the CRBRP, (2) place those identified risks in perspective with other societal risks, and (3) aid in determining whether accident risks from the CRBRP are comparable to those of previously licensed reactors. Achievement of the objectives of this study has required identification of significant contributors to risk in a logical

and orderly manner. Consideration of a comprehensive set of accident initiators, inclusion of experience data with a conservative bias, reliance upon proven methods and techniques, evaluation of a wide range of radionuclide releases and associated health effects, and utilization of experienced risk analysts are the salient elements employed in the systematic approach to this study. This together with heavy reliance on experience gained during years of Light Water Reactor (LWR) design, licensing, and operation provides reasonable assurance that the study objectives have been achieved. Results of the CRBRP Safety Study indicate that the risk arising from the operation of CRBRP is small in comparison to other local societal risks, and that CRBRP risk is comparable to the risk from previously licensed nuclear power plants as identified in the Reactor Safety Study.

- 992 (CS-INIS--199) ACCIDENT OF SODIUM STEAM GENERATOR ELEMENT - LEWISA PROGRAM. Korinek, J.; Stehlik, J.; Ricany, J.; Simerka, I.; Bucko, S.; Krchnakova, M. (Vyskumny Ustav Energeticky, Bratislava (Czechoslovakia)). 22 Dec 1975. 30p. (In Czech). Vyskumny Ustav Energeticky, Bratislava, Czechoslovakia.

The equation of the liquid motion in an unbranched tube and its numerical solution for the motion of a sodium and water column during an accident of a modular sodium steam generator is given. The LEWISA computer code mathematically modelling this accident, the input data of an exemplary accident and calculated values acquired by means of the code are described.

- 993 (CS-INIS--201) ANALYSIS OF FUEL-COOLANT INTERACTION IN THE INITIATION AND POSTEXPLOSIVE STAGES OF FAST REACTOR CREDIBLE ACCIDENTS. Suchomel, J.; Adamik, V.; Misak, J.; Zajic, A.; Simerka, I.; Sprusanska, M.; Polakova, Z.; Bazikova, V.; Stehlikova, Z. (Vyskumny Ustav Energeticky, Bratislava (Czechoslovakia)). 27 Feb 1976. 40p. (In Czech). Vyskumny Ustav Energeticky, Bratislava, Czechoslovakia.

The results of molten fuel-sodium interaction analysis are presented. The INTER code points out the heat transfer, the CEFRA code the influence on the process of structure dynamics. During the liquid expansion phase of MFCI in a tough tube, a pressure pulse of 1000 bar shorter than 1 ms is generated; the two-phase mixture subsequently expands from hundreds to units of bars. Due to the elasticity of the tube wall, pressure oscillations in the liquid phase arise with a lower amplitude. Thermal to mechanical energy conversion in the postexplosive MFCI in an RER-600 project was calculated for the determination of its vessel and cover loading.

- 994 (CS-INIS--212, pp 86-100) CONSTITUTION AND PROPERTIES OF CRMONB STEELS FOR STEAM GENERATOR COMPONENTS OF SODIUM-COOLED NUCLEAR REACTORS. Walder, V.; Gottwald, M. (Vyzkumny Ustav Hutnictvi Zeleza, Dobru (Czechoslovakia)). [nd].

From 5. international symposium on heat-resistant metal materials; Vsetin, Czechoslovakia (13 Sep 1976).

$NbC_{sub}(x)N_{sub}(y)$ ($x+y=1$) where x and y are functions of the chemical composition of steel and of precipitation temperature is the main precipitating phase in CrMoNb steel. A brief description is given of precipitation phase behaviour. Heat treatment affects mechanical properties dependent mainly on the size and distribution of precipitating particles. The amount of insoluble precipitates is affected by

austenitization temperature and time. Melt containing 1.11% Nb was less sensitive to austenitization temperature increase. The cooling rate also affects mechanical properties. In steels with excess of Nb malleabilizing is accompanied with Fe₂Nb precipitation increasing the precipitation strengthening of material at low malleabilizing temperatures. At a precipitation temperature of 750 degC or above coarse ferritic grains were observed.

- 995 (CSNI-R--14, pp Paper 12) INSERVICE INSPECTION OF PFR SECONDARY HEAT EXCHANGERS. Hudgell, R.J. (UKAEA, Risley. Nuclear Power Development Labs.). 1978.

From Specialists meeting on the ultrasonic inspection of reactor components; Daresbury, UK (27 Sep 1976).

In-service inspection of PFR secondary heat exchangers requires, for each tube bundle, systems capable of detecting defects in both the bore and outer surface of the tubes; specifically, pitting and cracking in the bore surface and fretting and thinning on the outer. Full 100% inspection of bends and straight portions is desirable together with an examination of the tube plate and in particular the tube/tube plate weld region. The current state of development to meet these requirements is described. An ultrasonic method of inspecting the tube plate for cracks which might extend into the tube plate is described, and also an eddy current method for detecting surface breaking cracks in the bores of the tube plate holes.(UK).

- 996 (CSNI-R--14, pp Paper 2) REVIEW OF THE PROCEEDINGS OF THE 8TH WORLD NDT CONFERENCE, ANNES, SEPTEMBER 1976. Hudgell, R.J. (UKAEA, Risley. Nuclear Power Development Labs.). 1978.

From Specialists meeting on the ultrasonic inspection of reactor components; Daresbury, UK (27 Sep 1976).

Among the papers given at the 8th World NDT Conference, which are reviewed, were several on light water reactors. One covered the development of equipment for in-service inspection, another discussed advances in electronics, transducer design, mechanical scanning, data presentation and evaluation of results in NDT, especially for nuclear vessels, and a third described an electronic system for ultrasonic inspection of reactor vessels and primary components. Of the remaining papers several dealt with acoustic emission related to nuclear inspection problems. One paper, only concerned inspection of heavy water reactors. It described an ultrasonic system for testing zirconium alloy pressure tubes using plate waves. Inspection of Rapsodie following the release of fluorine from concrete blocks surrounding the vessel was the only paper on in-service inspection of fast-reactors. Reference is made to several other topics which were discussed at the conference, including inspection of anisotropic material, twin crystal probes and their use for testing stainless steel welds.

- 997 (CSNI-R--14, pp Paper 10) ULTRASONIC INSPECTION DEVELOPMENT AT HEDL. Day, C.K.; Mech, S.J.; Michaels, T.E.; Dixon, N.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978.

From Specialists meeting on the ultrasonic inspection of reactor components; Daresbury, UK (27 Sep 1976).

Ultrasonic testing methods and equipment are being developed to support preservice and in-

service inspection of selected FFTF welds. A digital computer system is employed in the analysis of both simulated FFTF pipe sections and plate specimens containing fatigue cracks. It is anticipated that test evaluation standards containing fatigue cracks will partially eliminate questions formerly associated with weld test calibration producers by providing natural cracks which follow grain boundaries and stress patterns resembling piping situ conditions. Studies have revealed that commercial transducers may satisfy LMFBF ultrasonic pipe inspection applications.

- 998 (CSNI-R--32(Vol.2), pp 23.1-23.11) ELASTOPLASTIC FRACTURE MECHANICS APPROACH FOR LMFBF. Fortmann, M.; Link, F.; Zeibig, H. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). 1978.

From Safety of nuclear installations specialist meeting on elastoplastic fracture mechanics; Daresbury, UK (22 May 1978).

The structural materials used for LMFBF are austenitic steels from Type AISI 304, 316, or steels which are similar. Investigations show that a leak-before-break concept can be established for LMFBF piping systems and components, due to the ductile behaviour of these steels and operating conditions. A major element of this concept is the knowledge of fatigue crack growth patterns. The experiments which were carried out show that initial flaws will only grow slowly during the lifetime. The investigations should be extended into the plastic range. A test facility is under construction, where components such as nozzles, elbows and bellows can be tested under operational conditions with specified flaws in stress concentration areas.

- 999 (CURL--53) SYSTEM MODELING FOR TRANSIENT ANALYSIS OF LOOP-TYPE LIQUID-METAL-COOLED FAST BREEDER REACTORS. Khatib-Rahbar, M. (Cornell Univ., Ithaca, NY (USA). Nuclear Reactor Lab.). May 1978. 388p. Cornell Univ., Ithaca, NY.

Thesis.

A digital computer simulation model for the dynamic analysis of multiloop liquid-metal-cooled fast breeder reactor plants is developed. The model is capable of simulating operational transients, anticipated incidents, postulated accidents which do not disturb the system geometry, and a major pipe rupture accident in the primary heat transport system. The mathematical model includes: point reactor kinetics, reactivity feedback, and reactor power control; primary, secondary, and tertiary thermal and coolant dynamics governed by forced and natural circulation; and plant protection and control systems. A method is incorporated which calculates the negative inverse of the diagonal elements of the Jacobian matrix as the characteristic times for the 489 state variables modeling the entire system, and compares them with the preset integration timestep. A Runge-Kutta algorithm is applied to those state variables with slow and moderate time response, and a quasi-static approximation is applied to those with rapid response; i.e., the "stiff" equations. This assures numerical stability and is shown to greatly reduce the computation time requirements without much sacrifice in accuracy. The quasi-static equations are also utilized to determine the pre-transient steady state conditions. The system response to several transients has been calculated and compared to parallel calculations using the FORE-II and DEMO simulation models; these include: a seismic induced reactivity insertion, a spurious pump

trip, and a complete loss-of-forced-cooling leading to buoyancy induced natural circulation.

- 1000 (DOE/ER--0025-D) NUCLEAR STRATEGY OF THE DEPARTMENT OF ENERGY. (Department of Energy, Washington, DC (USA). Office of Energy Research). Apr 1979. 92p. Dep. NTIS, PC A05/MF A01.

An analysis of the important technical and economic parameters related to the timing and future deployment of nuclear power in the US is presented. The analysis is used first to form the basis of a set of alternative nuclear strategies and then to assist in a critical evaluation of these strategies. The strategies involve major decisions on breeder commercialization, fuel cycle development, and strengthening the nuclear option in the US. DOE energy projections indicate that nuclear power must play a substantial role in meeting future US energy needs, even under assumptions that are optimistic with respect to energy conservation and the increased use of coal-fired generating plants. Also, estimates indicate that nuclear power can have a substantial cost advantage in certain areas of the US. However, these estimates involve significant uncertainties, such as the relative environmental impact of the nuclear and coal cycles and the costs of meeting new environmental performance standards. There are three main elements in the fission portfolio: programs which extend the uranium resource base or which add to the efficiency of its use; breeder programs; and enabling programs.

- 1001 (DOE/ER--0054) PRELIMINARY COMPARATIVE ASSESSMENT OF LAND USE FOR THE SATELLITE POWER SYSTEM (SPS) AND ALTERNATIVE ELECTRIC ENERGY TECHNOLOGIES. Newsom, D.E.; Wolsko, T.D. (Argonne National Lab., IL (USA)). Apr 1980. Contract W-31-109-ENG-38. 26p. Dep. NTIS, PC A03/MF A01.

A preliminary comparative assessment of land use for the satellite power system (SPS), other solar technologies, and alternative electric energy technologies has been conducted. The alternative technologies are coal-gasification/combined-cycle, coal fluidized-bed combustion (FBC), light water reactor (LWR), liquid metal fast breeder reactor (LMFBR), terrestrial photovoltaics (TPV), solar thermal electric (STE), and ocean thermal energy conversion (OTEC). Fusion was not included in this preliminary work but will be a part of the final evaluation based on available research, to identify a suitable assessment methodology, and to identify data deficiencies. The major issues of a land use assessment are the quantity, purpose, duration, location, and costs of the required land use. The phased methodology described treats the first four issues, but not the costs. Several past efforts at comparative or single-technology assessment are reviewed briefly. The current state of knowledge about land use is described for each technology. Conclusions are drawn regarding deficiencies in the data on comparative land use and needs for further research. (WHK)

- 1002 (DOE/ET--0009) COMPENDIUM OF COMPUTER CODES FOR THE SAFETY ANALYSIS OF FAST BREEDER REACTORS. (Department of Energy, Washington, DC (USA). Div. of Reactor Research and Development). Oct 1977. 432p. Dep. NTIS, PC A14/MF A01.

The objective of the compendium is to provide the reader with a guide which briefly describes many of the computer codes used for liquid metal fast breeder reactor safety

analyses, since it is for this system that most of the codes have been developed. The compendium is designed to address the following frequently asked questions from individuals in licensing and research and development activities: (1) What does the code do. (2) To what safety problems has it been applied. (3) What are the code's limitations. (4) What is being done to remove these limitations. (5) How does the code compare with experimental observations and other code predictions. (6) What reference documents are available.

- 1003 (DOE/ET--0050/1) PERFORMANCE MEASUREMENT SYSTEM: SYSTEM DESCRIPTION. (Department of Energy, Washington, DC (USA). Div. of Reactor Research and Technology). 28 Jan 1977. 27p. (RDD/PMS--1). Dep. NTIS, PC A03/MF A01.

The summary description describes the Performance Measurement System (PMS) and Mini-Performance Measurement System (Mini-PMS) used by the Reactor Development and Demonstration Division (RDD) on selected contracts within Division projects. The intent of the pamphlet is to provide a summary description of the two systems and their operation for the benefit of ERDA and industry personnel. As its name implies, Mini-PMS is a derivative of PMS. Basically, PMS and Mini-PMS differ only in that Mini-PMS is a less rigid version of PMS. The general differences between PMS and Mini-PMS in the areas of their application, implementation, and operation are listed and discussed.

- 1004 (DOE/ET--0050/2) PERFORMANCE MEASUREMENT SYSTEM IMPLEMENTATION GUIDE. RRT MANAGEMENT CONTROL SYSTEM. (Department of Energy, Washington, DC (USA). Div. of Reactor Research and Technology). Apr 1978. 113p. (RDD/PMS--2). Dep. NTIS, PC A06/MF A01.

Revision of February 4, 1978.

The document provides uniform guidance for the Reactor Development and Demonstration Division (RDD) and other Energy Research and Development Administration (ERDA) organizations responsible for reviewing and monitoring contractor systems complying with the criteria for cost and schedule control systems of the RDD Performance Measurement System (PMS). Detailed PMS surveillance guidance is provided in RDD/PMS-5, Performance Management System Surveillance Guide. Specific PMS responsibilities of RDD and other participating organizations are stated in RDD Policy and Procedural Guide No. 26, Application and Implementation of PMS. For the purpose of this guide the term 'criteria' is synonymous with the PMS criteria for cost and schedule control systems. The term 'performance' is intended to mean 'cost/schedule performance.'

- 1005 (DOE/ET--0050/3(Rev.)) PERFORMANCE MEASUREMENT SYSTEM: CONTRACTOR REPORTING INSTRUCTIONS. (Department of Energy, Washington, DC (USA). Div. of Reactor Research and Technology). 14 Jan 1977. 55p. (RDD/PMS--3(Rev.)). Dep. NTIS, PC A04/MF A01.

The document provides guidance which has been approved by the Director of the Reactor Development and Demonstration Division (D/RDD) for use in preparation of the Management Control System Description, the Summary System Description, the Work Breakdown Structure Dictionary, the Cost Performance Report (CPR), the Cost/Schedule Status Report (C/SSR), the Schedule Status Report (SSR), and the Contract Funds Status Report (CFSR) of the Performance Measurement System (PMS) and the Mini-PMS.

1006 (DOE/ET--0050/4) PERFORMANCE MEASUREMENT SYSTEM: DATA ANALYSIS GUIDE. (Department of Energy, Washington, DC (USA). Div. of Reactor Research and Technology). 30 Mar 1976. 80p. (RDD/PMS--4). Dep. NTIS, PC A05/MF A01.

This guide provides a framework for using data generated by a management control system that complies with the requirements of the RDD Performance Measurement System (PMS). It was developed to assist ERDA Managers in their roles as decision makers. It presents: (1) types of data that are available, (2) various data analysis techniques applicable, (3) depiction of the data, and (4) benefits from use of the data.

1007 (DOE/ET--0050/5) PERFORMANCE MEASUREMENT SYSTEM: SURVEILLANCE GUIDE. (Department of Energy, Washington, DC (USA). Div. of Reactor Research and Technology). 30 Mar 1976. 58p. (RDD/PMS--5). Dep. NTIS, PC A04/MF A01.

The guide provides uniform guidance for the Reactor Development and Demonstration Division (RDD) and other Energy Research and Development Administration (ERDA) organizations responsible for surveillance of the RDD Performance Measurement System (PMS) criteria for cost and schedule control systems. Guidance for implementation of the PMS criteria and acceptance of a contractor's management control system is provided in RDD/PMS-2, PMS Implementation Guide. The procedural guidance contained in the guide should be adapted to specific organizational structures and to specific situations as they arise. Details concerning the PMS surveillance of each contractor's management control system should be developed by the Cognizant Contracting Office (CCO), coordinated with the Cognizant Federal Auditor (CFA), and agreed upon by RDD Project Management. Any procedures employed should be consistent with the guidance contained herein and in the PMS Implementation Guide. The guidance furnished will provide assistance to: (1) RDD Project Management in formulating surveillance plans and determining resource requirements; (2) RDD surveillance personnel in accomplishing their surveillance functions; and (3) contractors in understanding the requirements of the surveillance program and responding to them.

1008 (DOE/ET--0050/6) PERFORMANCE MEASUREMENT SYSTEM: WORK BREAKDOWN STRUCTURE (WBS) GUIDE. (Department of Energy, Washington, DC (USA). Div. of Reactor Research and Technology). Apr 1978. vp. (RDD/PMS--6).

The guide provides guidance on the use of the Work Breakdown Structure (WBS) technique to Reactor Development and Demonstration Division (RDD) managers responsible for projects supporting the LMFBR Program. The guide covers the types of WBSs, how and by whom they are prepared, and how they should be used by RDD and contractors for effective management and control.

1009 (DOE/ET--0050/7) PERFORMANCE MEASUREMENT SYSTEM: MINI-PMS GUIDE. (Department of Energy, Washington, DC (USA). Div. of Reactor Research and Technology). 14 Jan 1977. 72p. (RDD/PMS--7). Dep. NTIS, PC A04/MF A01.

This guide provides guidance to Reactor Development and Demonstration Project Management and contractors on the use of the Mini-Performance Measurement System (Mini-PMS) as a contract management tool. As its name

implies, Mini-PMS is a derivative of the RDD Performance Measurement System (PMS) applied to selected contracts/subcontracts within major RDD projects. Mini-PMS differs from PMS in that Mini-PMS does not require that contractors' management control systems comply with a set of criteria for such systems and offers a more flexible approach to contract performance reporting. Mini-PMS is applied to contracts which require visibility and control in order to meet the contract and related project objectives. In addition, contractors should consider applying Mini-PMS to their subcontracts. Mini-PMS supports the Project Management functions through a series of interrelated RDD/contractor activities that include: (1) development and use of work breakdown structures (WBSs) that define and organize the project/contract work; (2) use of contractor management control systems that facilitate performance measurement within the WBS framework; and (3) use of a reporting system based on Mini-PMS data elements to report contract status. When combined, these activities provide the means for correlating the contract's cost, schedule, and technical accomplishments.

1010 (DOE/ET--0050/8) PERFORMANCE MEASUREMENT SYSTEM REVIEW TEAM HANDBOOK. RRT MANAGEMENT CONTROL SYSTEM. (Department of Energy, Washington, DC (USA). Div. of Reactor Research and Technology). 30 Mar 1977. 116p. (RDD/PMS--8). Dep. NTIS, PC A06/MF A01.

The purpose of the handbook is to provide guidance and reference material for personnel selected to participate in PMS/Mini-PMS implementation activities. It includes procedures and evaluation techniques which have been developed to assist in timely and effective evaluation of a contractor's implementation of the RDD contract performance measurement requirements.

1011 (DOE/ET/37203--1) EXPERIMENTAL EVALUATION OF ACOUSTIC AGGLOMERATORS AS AN AIR CLEANING SYSTEM CONCEPT FOR EMERGENCY USE IN LMFBR PLANTS. FINAL REPORT. Shaw, D.T.; Wegrzyn, J. (State Univ. of New York, Buffalo (USA). Lab. for Power and Environmental Studies). Nov 1979. Contract EN-77-S-02-4260. 52p. Dep. NTIS, PC A04/MF A01.

In the development of the Liquid Metal Fast Breeder Reactor, a major safety problem concerns the suppression of the sodium oxides aerosol particles produced during a hypothetical core disruptive accident. Some theoretical calculations are presented concerning sonic agglomeration rates of such a system in terms of acoustic frequency, intensity and particles. The information is useful in providing some guidelines for the evaluation and design of sonic systems for LMFBR emergency applications.

1012 (DOE/ET/37205--T1) CHEMICAL SIMULATION MODELING OF HEAT TRANSFER IN LMFBRs. Wiberg, D.V. (Brigham Young Univ., Provo, UT (USA). Dept. of Chemical Engineering). Apr 1980. Contract AC02-76ET37205. 126p. NTIS, PC A07/MF A01.

Thesis.

A chemical reaction conducted in a wax cylinder was used in this study to simulate a CDA occurring in a reactor. The exothermic reaction of dimethyl sulfoxide (DMSO) with acetyl chloride (AC) was used as the chemical reaction to simulate the internally heated liquid pool. SUN-3420 wax was used to construct the cylindrical reaction vessel to simulate

cladding and containment.

- 1013 (DOE/EV--0046(Vol.2), pp 111-142) COMBUSTION OF SODIUM IN THE OPEN ATMOSPHERE. Morewitz, H.A.; Johnson, R.P.; Nelson, C.T.; Otter, J.M. (Rockwell Hanford Operations, Richland, WA). Sep 1979.

From Environmental control symposium; Washington, DC, USA (28 Nov 1978).

A series of sodium fire tests has been conducted in ambient air at a meteorological test site. This test series was designed to simulate hypothetical accidents which might occur in the heat transport system of an LMFBR. Measurements of concentration, agglomeration, fallout, and chemical species of sodium combustion products were made as a function of downwind distance. In each of the first two tests, approx. 23 kg of 540°C sodium was sprayed as a fan of 250-μm sodium drops across the wind, from heights of 5 or 6 m. Each release took a few minutes. Two additional tests were performed by releasing sodium through 9.5-mm diameter jets at a height of 30 m. In each test, the sodium jet was aimed horizontally across the wind, and followed a downward parabolic trajectory, releasing burning sodium drops along its track. In a third type of test, sodium was burned for 60 min as a pool in a 1.5 m² burn pan at 9 m/s wind velocity. Approximately 30% of the combustion products became airborne. Large agglomerates fell out as they moved downward, depositing 2 kg/m² at 1 m downwind from the edge of the pan.

- 1014 (DOE/NE--0001(Vol.6)) NUCLEAR PROLIFERATION AND CIVILIAN NUCLEAR POWER: REPORT OF THE NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM. VOLUME VI. SAFETY AND ENVIRONMENTAL CONSIDERATIONS FOR LICENSING. (Department of Energy, Washington, DC (USA)). Dec 1979. 139p. Dep. NTIS, PC A07/MF A01.

Volume 6 of the Nonproliferation Alternative Systems Assessment Program report addresses safety and environmental considerations in licensing the principal alternative nuclear reactors and fuel cycles in the United States for large-scale commercial nuclear power plants. In addition, this volume examines the safety and environmental considerations for licensing fuel service centers. These centers, which have been proposed for controlling sensitive fuel-cycle facilities and special nuclear materials, would contain a combination of such facilities as reprocessing plants, fabrication plants, and reactors. For this analysis, two fuel service center concepts were selected - one with power-generating capability and one without. This volume also provides estimates of the time required for development of large-scale commercial reactor systems to reach the construction permit application stage and for fuel-cycle facilities to reach the operating license application stage, which is a measure of the relative technical status of alternative nuclear systems.

- 1015 (DOE/NE--0003/6) LIQUID-METAL FAST-BREEDER REACTORS: PRELIMINARY SAFETY AND ENVIRONMENTAL INFORMATION DOCUMENT. VOLUME VI. (Department of Energy, Washington, DC (USA)). Jan 1980. 175p. Dep. NTIS, PC A08/MF A01.

Information is presented concerning LMFBR design characteristics; uranium-plutonium/uranium recycle homogeneous core; uranium-plutonium/uranium spiked recycle heterogeneous core; uranium-plutonium/uranium spiked recycle homogeneous core; uranium-plutonium/thorium

spiked recycle heterogeneous core; uranium-plutonium/thorium spiked recycle homogeneous core; thorium-plutonium/thorium spiked recycle homogeneous core; denatured uranium-233/thorium cycle homogeneous core; safety consideration for the LMFBR; and environmental considerations.

- 1016 (DOE/TIC--10063) STRUCTURAL RESPONSE OF 1/20-SCALE MODELS OF THE CLINCH RIVER BREEDER REACTOR TO A SIMULATED HYPOTHETICAL CORE DISRUPTIVE ACCIDENT. TECHNICAL REPORT 4. Romander, C.M.; Cagliostro, D.J. (SRI International, Menlo Park, CA (USA)). Oct 1978. Contract EY-76-C-03-0115-097. 266p. Dep. NTIS, PC A12/MF A01.

Five experiments were performed to help evaluate the structural integrity of the reactor vessel and head design and to verify code predictions. In the first experiment (SM 1), a detailed model of the head was loaded statically to determine its stiffness. In the remaining four experiments (SM 2 to SM 5), models of the vessel and head were loaded dynamically under a simulated 661 MW=sec hypothetical core disruptive accident (HCDA). Models SM 2 to SM 4, each of increasing complexity, systematically showed the effects of upper internals structures, a thermal liner, core support platform, and torospherical bottom on vessel response. Model SM 5, identical to SM 4 but more heavily instrumented, demonstrated experimental reproducibility and provided more comprehensive data. The models consisted of a Ni 200 vessel and core barrel, a head with shielding and simulated component masses, an upper internals structure (UIS), and, in the more complex models SM 4 and SM 5, a Vi 200 thermal liner and core support structure. Water simulated the liquid sodium coolant and a low-density explosive simulated the HCDA loads.

- 1017 (DOE/TIC--10099) FRESS PIN FAILURE MODEL AND ITS APPLICATION TO E-8 TREAT TEST. Kalimullah. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 19p. Dep. NTIS, PC A02/MF A01.

FRESS is a cladding rupture prediction model for an irradiated mixed-oxide LMFBR fuel pin during transient heating based only on the internal pressurization of the cladding by the fission gas released from fuel grains during the transient. The model is applied to the analysis of the hottest PNL-10-53 pin in the 7-pin E-8 TREAT test which simulates a \$3/sec transient overpower. Although the uncertainties of the inputs to the temperature calculation done with the COBRA code have not been included, the uncertain input parameters to FRESS have been varied over their estimated uncertainties. The cladding rupture predictions are a few tens of milliseconds late compared to the most probable failure time detected in the test. However, these calculations seem to indicate that fission gas pressure is a significant mechanism for causing clad rupture in this test.

- 1018 (DOE/TIC--10100) ANALYSIS OF A PROTOTYPIC AND A SIMULANT-EXPERIMENT RELATED TO FREEZING IN FLOW CHANNELS. Hakim, S.J. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 23p. Dep. NTIS, PC A02/MF A01.

An analysis is presented of two experiments, one involving simulant materials, and a second more prototypic experiment, related to freezing of molten LMFBR core material in flow channels. Good agreement is obtained between predictions and experimental results for both experiments. It is argued that simulant experiments could

provide a reasonable vehicle for validation of isolated phenomena related to the freezing process while prototypic experiments provide a more realistic view of the integrated behavior of molten core material during the accident.

- 1019 (DOE/TIC--10101) ANALYSIS OF TREAT TRANSIENT OVERPOWER EXPERIMENTS USING THE PLUTO2 CODES. Wider, H.U.; Semenza, L.A. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 12p. Dep. NTIS, PC A02/MF A01.

Pre- and post-test computer analyses of TREAT transient overpower test H6 are presented and compared with an earlier analysis of E8. The computer models used were the PLUTO code and its successor, the PLUTO2 code. A good agreement with several events could be achieved in the post test simulations, which supports the basic modeling in the codes used. The pre-test analyses suffered from the uncertainty in the pin failure sequence. Fission-gas pressures in the pins at failure time were on the order of 15-30 MPa in both tests. Plenum fission-gas release may have played an important role in both tests.

- 1020 (DOE/TIC--10102) ANALYSIS OF TREAT TESTS L7 AND L8 WITH SAS3D, LEVITATE AND PLUTO2. Bowers, C.H.; Tentner, A.M.; Wider, H.U. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 17p. Dep. NTIS, PC A02/MF A01.

Fuel motion in TREAT Tests L7 and L8 is analyzed with the SAS3D, LEVITATE, and PLUTO2 computer codes. These results are compared with fuel motion data obtained from the fast neutron hodoscope. It is concluded that fission gas acts to disperse fuel during the early stages of fuel motion.

- 1021 (DOE/TIC--10187) FABRICATION DATA FOR THE US/UK SAFETY TEST PROGRAM L01 CALIBRATION FUEL PINS. Steffen, J.M.; Manuel, J.L. (comps.). (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1979. Contract EY-76-C-14-2176. 36p. AT.

Data are presented that relate to the fabrication of fuel pins for the LMFBR safety test program.

- 1022 (DOE/TIC--10188) DATA FOR PNC FUEL PINS FOR HEDL P-E01 TEST. Steffen, J.M.; Manuel, J.L. (comps.). (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1979. Contract EY-76-C-14-2176. 37p. AT.

Fabrication and quality assurance data are presented for the LMFBR test fuel pins.

- 1023 (DOE/TIC--10224) LARGE LIQUID METAL FAST BREEDER REACTOR INTERMEDIATE HEAT EXCHANGER MAINTENANCE STUDY REPORT. Johnson, R.H. (Foster Wheeler Corp., Livingston, NJ (USA)). 8 Dec 1977. Contract W-31-109-ENG-38. 38p. AT.

Current designs for Intermediate Heat Exchangers (IHX) planned for use in Commercial LMFBR systems call for a straight tube configuration without a central downcomer. The maintenance of such a unit involves three (3) major operations: detection of leaking tubes; location of defective tubes; and plugging of leaking tubes. Two of the main objectives of this study were to investigate IHX maintainability in the areas of tube leak detection and tube plugging.

- 1024 (DOE/TIC--10240, pp 24p, Attachment 2)

APPLICATION OF ALLOY 718 IN THE UPPER INTERNALS STRUCTURE OF THE CLINCH RIVER BREEDER REACTOR. Schmidt, J.E. (Westinghouse Electric Corp., Madison, PA). 1975.

Status of technology of Alloy 718 for nuclear applications. Minutes of meeting, October 30, 1975.

After a description of the Clinch River Breeder Reactor, the utilization of Inconel 718 is described and research needed is explained. (FS)

- 1025 (DOE/TIC--10240) STATUS OF TECHNOLOGY OF ALLOY 718 FOR NUCLEAR APPLICATIONS. MINUTES OF MEETING, OCTOBER 30, 1975. Roche, T.K.; Brinkman, C.R. (comps.). (Oak Ridge National Lab., TN (USA)). 1975. Contract W-7405-ENG-26. 302p. Dep. NTIS, PC A14/MF A01.

The objective of the Alloy 718 meeting held at ORNL on October 30, 1975 were to evaluate the status of the technology of the alloy. Subjects presented included melting, fabrication, heat treatments, welding techniques, weldment properties, product from availability, use in the nuclear industry, design data, code requirements, and mechanical and physical properties. Each paper was processed separately for the Energy Data Base. (FS)

- 1026 (DOE/TIC--10240, pp 100p, Attachment 3) EVALUATION OF THE ADDITIONAL MATERIALS DATA NEEDED TO SUPPORT THE USE OF ALLOY 718 IN NON-WELDED NUCLEAR REACTOR APPLICATIONS. Snow, A. (Westinghouse Electric Corp., Madison, PA). 1975.

Status of technology of Alloy 718 for nuclear applications. Minutes of meeting, October 30, 1975.

An evaluation is made of data needed to support the use of Alloy 718 in nonwelded nuclear reactor applications. Included are low and elevated temperature tensile and fatigue data and test results, physical property information, and plasticity and aging effects. (FS)

- 1027 (DOE/TIC--10240, pp 10p, Attachment 4) MATERIAL PROPERTIES REQUIREMENTS AND PRIORITIES MATRIX FOR CRBRP. Bitner, J.L. (Westinghouse Electric Corp., Madison, PA). 1975.

Status of technology of Alloy 718 for nuclear applications. Minutes of meeting, October 30, 1975.

The latest material properties requirements, and priorities matrix for CRBR are tabulated. Properties include strength, residual ductility, physical properties, fatigue, creep, fracture, radiation effects, aging and corrosion. (FS)

- 1028 (DOE/TIC--10240, pp 11p, Attachment 6) ALLOY 718 UTILIZATION IN THE FAST FLUX TEST FACILITY; ALLOY 718 DESIGN DATA CURRENTLY IN NUCLEAR SYSTEMS MATERIALS HANDBOOK; GENERAL AND SPECIFIC DESIGN DATA NEEDS FOR ALLOY 718; ALLOY 718 FOR CODE CASE 1592-4 APPLICATIONS (BOLTING); GUIDELINES FOR REQUIRED DATA ON NEW MATERIALS FOR SECTION III, ELEVATED TEMPERATURE NUCLEAR CONSTRUCTION. Moen, R.A. (Westinghouse Hanford Co., Richland, WA). 1975.

Status of technology of Alloy 718 for nuclear applications. Minutes of meeting, October 30, 1975.

Information is presented concerning Inconel 718 utilization in the Fast Flux Test Facility, design data currently in Nuclear Systems Materials Handbook, general and specific design

data needs,, bolting applications and guidelines for required data. (FS)

- 1029 (DOE/TIC--11038) SODIUM HYDRIDE PRECIPITATION IN SODIUM COLD TRAPS. McPheeters, C.C.; Raue, D.J. (Argonne National Lab., IL (USA)). Oct 1979. Contract W-31-109-ENG-38. 29p. Dep. NTIS, PC A03/MF A01.

A series of experiments have been performed to test a calculational model for precipitation of NaH in sodium cold traps. The calculational model, called ACTMODEL, is a computer simulation that uses the system geometry and operating conditions as input to calculate a mass transfer coefficient and the distribution of NaH in a cold trap. The ACTMODEL was tested using an analytical cold trap (ACT) that is simple and essentially one-dimensional. The ACT flow and temperature profile may be controlled at any desired condition. The ACT was analyzed destructively after each test to measure the actual NaH distribution. Excellent agreement was obtained between the ACTMODEL simulations and the experiments.

- 1030 (DOE/TIC--11039) EXPERIMENTS ON COLD TRAP REGENERATION BY NAH DECOMPOSITION. McPheeters, C.C.; Skladzien, S.B.; Raue, D.J. (Argonne National Lab., IL (USA)). Oct 1979. Contract W-31-109-ENG-38. 23p. Dep. NTIS, PC A02/MF A01.

Cold trap regeneration may be very important in future LMFBRs because of the expected high hydrogen source from the steam generators. This hydrogen precipitates as NaH in the cold trap and may fill the trap within one year of operation. Several methods of cold trap regeneration were considered, but the simplest and least expensive appears to be decomposition of NaH under vacuum at elevated temperatures. Experiments were done to assess the feasibility of this method for cold trap regeneration. Small-scale simulated cold traps (SCT) were located with NaH and NaH plus Na₂O, and were heated both under vacuum and under a sweep gas at 100 kPa. The evolved hydrogen was converted to water by a CuO bed and collected in a weighting tube.

- 1031 (DOE/TIC--11040) COLD TRAP REQUIREMENTS FOR PHTS AND IHTS SODIUM PURIFICATION. Sletten, H.L. (Atomics International Div., Canoga Park, CA (USA)). 31 May 1974. Contract EY-76-C-03-0824-025. 76p. (TI--099-811-013). Dep. NTIS, PC A05/MF A01.

Cold trap requirements for the Primary Heat Transfer System (PHTS) and the Intermediate Heat Transfer System (IHTS) are defined. The bases for these requirements are (1) maintaining concentrations of oxygen and hydrogen within given limits, (2) controlling the transport of tritium by coprecipitating it in cold traps together with hydrogen, and (3) providing the necessary volume capacity for collecting impurities. It is concluded that the 60 gpm, FFTF Type I, NaK cooled, cold trap will satisfy the PHTS requirements. For each of the IHTS loops a single, 60 gpm, cold trap is required; and an aircooled trap similar to the FFTF Type I design except for a 3-foot longer crystallizer section has been chosen for this service.

- 1032 (DOE/TIC--11041) THERMAL AND HYDRAULIC ANALYSES OF THE SYSTEM 81 COLD TRAPS. Kim, K. (Atomics International Div., Canoga Park, CA (USA)). 15 Jun 1977. Contract EY-76-C-03-0824-025. 174p. Dep. NTIS, PC A08/MF A01.

Portions of document are illegible.

Thermal and hydraulic analyses of the System 81 Type I and II cold traps were completed except for thermal transients analysis. Results are evaluated, discussed, and reported. Analytical models were developed to determine the physical dimensions of the cold traps and to predict the performance. The FFTF cold trap crystallizer performances were simulated using the thermal model. This simulation shows that the analytical model developed predicts reasonably conservative temperatures. Pressure drop and sodium residence time calculations indicate that the present design will meet the requirements specified in the E-Specification. Steady state temperature data for the critical regions were generated to assess the magnitude of the thermal stress.

- 1033 (DOE/TIC--11042) COMPUTERIZED FUEL MEASUREMENT FROM X-RAY RADIOGRAPHY. Green, D.R.; Karnesky, R.A.; Bromley, C. (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1979. Contract EY-76-C-14-2170. 11p. Dep. NTIS, PC A02/MF A01.

Techniques have recently been developed which allow nondestructive dimensional characterization of fuel pellets within fuel pins. Fuel pellet diameters can be measured with an accuracy of ± 0.0005 inch (0.0012 cm) using standard vendor-supplied radiography, or with an accuracy of ± 0.0002 inch (0.0005 cm) using x-ray radiography produced by a special HEDL developed technique. These data are needed for the detailed evaluation of the effects of irradiation on fuel behavior. The radiographs to be analyzed are transformed into a digital form using a computer controlled x-y scanning microdensitometer. The digital image is then recorded on a magnetic tape which is then sent to the central computer (CDC 6600) for analysis with FORTRAN IV software. Pellet diameters are determined, statistically analyzed, and the results are tabulated with a high speed line printer. These reports are also placed on magnetic tape storage for future computerized comparison and evaluation.

- 1034 (DOE/TIC--11060) SIEX DESIGN PREDICTIONS FOR THE PNC FUEL PINS IN THE HEDL P-E01 POWER-TO-MELT TEST. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 13p. Dep. NTIS, PC A02/MF A01.

During the design phase of the HEDL P-E01 power-to-melt test, a series of design predictions were generated for the three PNC pins using the SIEX fuel pin modeling code. This document tabulates a series of selected PNC pin design predictions as requested by M. Shinohara during his visit to HEDL (October 8-19, 1979).

- 1035 (DOE/TIC--11087) FFTF DRIVER FUEL PELLETS: TYPICAL PELLETT LOT DATA. (Babcock and Wilcox Co., Apollo, PA (USA). Nuclear Materials and Mfg. Div.). [nd]. Contract EY-76-C-02-3109. 13p. Dep. NTIS, PC A02/MF A01.

Quality assurance data for FFTF reactor fuel pellets are presented.

- 1036 (DOE/TIC--11088) FFTF DRIVER FUEL PELLETS: TYPICAL PELLETT LOT DATA. (Kerr-McGee Corp., Oklahoma City, OK (USA)). [nd]. 12p. Dep. NTIS, PC A02/MF A01.

Quality assurance data are presented for FFTF driver fuel pellets.

1037 (DOE/TIC--11114) LIQUID METAL ENGINEERING CENTER (LMCE) ANNUAL TECHNICAL PROGRESS REPORT, JANUARY-DECEMBER 1975. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 30 Apr 1976. Contract EY-76-C-03-0700. 22p. Dep. NTIS, PC A02/MF A01.

Portions of document are illegible.

During operation of the prototype FFTF sodium pump, it is necessary to know the position of the pump shaft relative to the sodium-lubricated hydrostatic bearing journals. Performance of the Kaman proximity measurement system was known to be affected by the type of media between the gap to be measured. Since Kaman Sciences Corp. did not have liquid sodium testing capabilities, LMCE was requested to evaluate a spare prototype system. Determination of the sodium performance characteristics of the Kaman proximity systems was a necessary requisite for use of four prototype systems in the FFTF prototype pump test program.

1038 (DOE/TIC--11115) THERMAL INSULATION SYSTEM DESIGN AND FABRICATION SPECIFICATION (NUCLEAR) FOR THE CLINCH RIVER BREEDER REACTOR PLANT. (Westinghouse Electric Corp., Oak Ridge, TN (USA). Advanced Reactors Div.). 21 Jul 1978. Contract EY-76-C-15-2395. 67p. Dep. NTIS, PC A04/MF A01.

This specification defines the design, analysis, fabrication, testing, shipping, and quality requirements of the Insulation System for the Clinch River Breeder Reactor Plant (CRBRP), near Oak Ridge, Tennessee. The Insulation System includes all supports, convection barriers, jacketing, insulation, penetrations, fasteners, or other insulation support material or devices required to insulate the piping and equipment cryogenic and other special applications excluded. Site storage, handling and installation of the Insulation System are under the cognizance of the Purchaser.

1039 (DOE/TIC--11116) CLINCH RIVER BREEDER REACTOR PLANT NUCLEAR INSULATION SYSTEM MATERIAL SELECTION REPORT. (Westinghouse Electric Corp., Pittsburgh, PA (USA). Advanced Energy Systems Div.). Oct 1977. Contract EY-76-C-15-2395. 40p. Dep. NTIS, PC A03/MF A01.

This insulation system material selection report presents a comparative evaluation of candidate insulation materials for use with the nuclear Steam Supply System (NSSS) of the Clinch River Breeder Reactor Plant (CRBRP). This evaluation takes a comprehensive view of the quantitative as well as qualitative characteristics of candidate insulation and protective covering materials.

1040 (DOE/TIC--11162) EBR-II OPERATING EXPERIENCE. Smith, C.R.F. (Argonne National Lab., IL (USA)). Jul 1978. Contract W-31-109-ENG-38. 220p. Dep. NTIS, PC A10/MF A01.

Operation of the EBR-2 reactor is presented concerning the performance of the heat removal system; reactor materials; fuel handling system; sodium purification and sampling system; cover-gas purification; plant diagnostics and instrumentation; recent improvements in identifying fission product sources in EBR-2; and EBR-2 safety.

1041 (DOE/TIC--11163) EBR-II PROJECT: RUN 100, MARCH 4 1979-APRIL 15, 1979. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 44p. Dep. NTIS, PC A03/MF A01.

Run 100 started on March 4 and ended on April 15, 1979, following 2139 Mwd of reactor operation. Reactor-operate interlocks and reactor-startup check sheets were completed on March 4. Criticality was reached at 2337 on March 4. The control rods were calibrated, and reactor power was increased, reaching 62.5 MWt at 0850 on March 5. Run 100A was to be a 48-hr irradiation of the flow-blockage-simulation experiment (X343). The FERD signal increased linearly with power and reached about 280 counts/s at 62.5 MWt.

1042 (DOE/TIC--11164) EBR-II PROJECT: RUN 99, JANUARY 3, 1979-FEBRUARY 22, 1979. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 49p. Dep. NTIS, PC A03/MF A01.

Run 99 was started on Jan. 9, 1979, and ended on Feb. 22, 1979, following 2344 Mwd of operation. The run was conducted in five segments because of fission-product releases. There were two scrams; one was caused by instrument malfunctions and one was a manual scram due to high FERD counts.

1043 (DOE/TIC--11165) EBR-II PROJECT: RUN 104, OCTOBER 7, 1979-DECEMBER 3, 1979. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 33p. Dep. NTIS, PC A03/MF A01.

Run 104 started on October 7, 1979, and ended on December 3, 1979, after 3106 Mwd of operation. Authorized power level was reduced from 62.5 MWt to 57-1/4 MWt as part of the Operational Safety Testing Program. The run was conducted in two segments because of a fission-product release. Two reactor scrams occurred during the run: one caused by failure of the latch clutch for control rod No. 1, and one caused by a low-flow trip for reactor coolant outlet.

1044 (DOE/TIC--11166) EBR-II PROJECT: RUN 101, APRIL 16, 1979-JUNE 25, 1979. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 53p. Dep. NTIS, PC A04/MF A01.

Run 101 started on April 18 and ended on June 25, 1979, after 2436 Mwd of operation. The run was conducted in two segments because of a fission-product release. There was one manual scram at the beginning of the run, before full power, because of high FERD count. Between the completion of run 100 on April 16 and the beginning of run 101, annual long-shutdown maintenance and scheduled plant modifications were in progress.

1045 (DOE/TIC--11167) EBR-II PROJECT: RUN 102, JUNE 25, 1979-AUGUST 20, 1979. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 38p. Dep. NTIS, PC A03/MF A01.

Run 102 started on July 4 and ended on August 20, 1979, following 2513 Mwd of reactor operation.

1046 (DOE/TIC--11168) EBR-II PROJECT: RUN 103, AUGUST 20, 1979-OCTOBER 6, 1979. (Argonne National Lab., IL (USA)). 1979. Contract W-31-109-ENG-38. 36p. Dep. NTIS, PC A03/MF A01.

Run 103 was started on August 27, 1979, and ended on October 6, 1979, following 2277 Mwd of operation. The run was conducted in two segments. There were no scrams.

1047 (DOUNREAY=TRANS--813) KALKAR NUCLEAR POWER STATION (SNR 300). (UKAEA Dounreay Nuclear Power Development Establishment). Aug 1978. 23p. British Library Lending Division, Boston Spa, Wetherby, West Yorkshire, England.

The report falls under the following headings: summary of the most important events in the first quarter of 1978; state of the planning (nuclear steam generating system; conventional sector); state of the safety technology and building regulation work (licensing processes); building progress report; deadline situation.

1048 (ECN--21) TRANSIENT TUBE BURST TESTS ON FAST REACTOR FUEL CLADDING. Kwast, H. (Stichting Energieonderzoek Centrum Nederland, Petten). Jul 1977. 80p. Dep. NTIS (US Sales Only), PC A05/MF A01.

Experimental results of transient tube burst tests on tubing of three different austenitic stainless steels, Werkstoff no. 1.4970 (Sandvik 12R72HV), 1.4981 and 1.4988 (DIN classification), are reported. The tubing geometry is representative for the canning of the sodium cooled fast reactor SNR-300. The internal tube pressure ranged from 30 to 120 Kg/cm². The heating rate was varied from 27 to 110degC/sec. Up to a heating rate of about 70degC/sec the failure temperatures can be reasonable well predicted by a secondary creep relation. The failure temperature of 1.4970 tubing is strongly influenced by the heating rate. This is much less the case for 1.4981 and 1.4988 tubing. The failure strains of the three tubings depend largely on failure temperature and except for material 1.4981, also on heating rate. Three different failure modes have been observed.

1049 (ECN--22) FAST REACTOR PROGRAMME. FIRST QUARTER 1977 PROGRESS REPORT. Hoekstra, E.K. (Stichting Energieonderzoek Centrum Nederland, Petten). Jul 1977. 50p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Results of central reactivity worth measurements on various fission product nuclides performed in STEK at Petten are compared with results of integral activation measurements performed on same in CFRMF at Idaho. Post-irradiation examination results are given of a burst fuel pin (R54-F36) fitted with a stainless steel DIN 1.4981 canning, and loss-of-cooling experiment R63-L14. The outcome of several thermal diffusivity measurements on 14 pellets of U(C,N) is reported. Results of low-cycle fatigue experiments on creep-conditioned specimens of stainless steel DIN 1.4948 are discussed. A study on the leakage of aerosol particles through cracks and ducts in a reactor containment wall is proposed. In the field of heat transfer and hydraulics, reports are given on a singlephase experiment using a thin-walled heater NANCY-054, Laser Doppler Anemometer measurements on water flowing through a square channel, temperature noise measurements, and a study of local boiling behind a flow blockage in a 28-rod bundle.

1050 (ECN--34) FAST REACTOR PROGRAM. COMBINED SECOND AND THIRD QUARTERS 1977 PROGRESS REPORT. Hoekstra, E.K. (Stichting Energieonderzoek Centrum Nederland, Petten). Jan 1978. 62p. Dep. NTIS (US Sales Only), PC A04/MF A01.

Average capture cross sections and their standard deviations of a number of fission product nuclides in an SNR-300 flux spectrum, derived from the integral data from STEK and CFRMF, are given. The HFR-TOP facility for

transient overpower experiments on fuel pins being irradiated in the pool-side facility for transient overpower experiments on fuel pins being irradiated in the pool-side facility of the HFR is described. The outcome of the thermal diffusivity measurements on U(C,N) pellets of different oxygen content is reported. Results of destructive tests on samples of stainless steel DIN 1.4948 used in the SNR-300 reactor vessel are discussed. Data on the floor of the SAUNA vessel to the covering sodium aerosol are given. Heat transfer and temperature noise analyses of the single and two-phase experiments performed behind a flow blockage in an electrically heated 28-rod bundle in a sodium coolant are reported.

1051 (ECN--41) FAST REACTOR PROGRAMME. FOURTH QUARTER 1977 PROGRESS REPORT. Hoekstra, E.K. (Stichting Energieonderzoek Centrum Nederland, Petten). May 1978. 67p. Dep. NTIS (US Sales Only), PC A04/MF A01.

Fast neutron capture cross section data of natural Mo are presented; a feasibility study on integral neutron spectrum measurements in the low-energy region of large LMFBR's is discussed. The post-irradiation examination results of the fuel pin of loss-of-cooling experiment R63-L18 and the progress made with preparations for the HFR-TOP transient overpower experiments on fuel pins under irradiation in the pool-side facility of the HFR are reported on. An exact determination of the thermochemical properties of uranyl halides is discussed. Results of mechanical tests on stainless steel DIN 1.4948 specimens are given. Aerosol leakages through blocks of concrete are determined. In the field of heat transfer and hydraulics, data on fully-developed turbulent flows in rod bundles have been obtained using the code VITESSE, results of Laser Doppler Anemometer measurements performed on water flowing through a square channel are given and progress is reported on the study of local boiling occurring behind a flow blockage in a 28-rod bundle.

1052 (ECN--48) FAST REACTOR PROGRAMME. FIRST QUARTER 1978 PROGRESS REPORT. Hoekstra, E.K. (Stichting Energieonderzoek Centrum Nederland, Petten). Sep 1978. 52p. Dep. NTIS (US Sales Only), PC A04/MF A01.

Recently evaluated neutron capture cross section data of ¹⁴²Nd, ¹⁴³Nd, ¹⁴⁴Nd, ¹⁴⁵Nd, ¹⁴⁶Nd, natural Nd and Mo as well as adjustment of the capture cross sections of ¹⁵²Sm and ¹⁵⁴Sm to fit integral data measured in STEK and CFRMF are discussed. The progress made with preparations for the HFR-TOP transient overpower experiments on fuel pins under irradiation in the pool-side facility of the HFR is reported on. Results are given of tensile tests on irradiated as well as on heat-treated stainless steel DIN 1.4948 specimens subjected to varying numbers of fatigue cycles. In the field of aerosol research, measured gas flow rates and pressure drops in stainless steel capillaries of various dimensions are compared with theory; the gas flow and aerosol penetration in cracks, artificially introduced in concrete test specimens, have been determined. Criteria in selecting the right light-scattering particles for use in Laser Doppler Anemometer measurements are given; the results of single and two-phase experiments with the second 28-rod bundle and the hydrodynamics during single-bubble boiling in the first bundle are discussed.

1053 (ECN--50) EFFECTS OF NEUTRON

IRRADIATION AND FATIGUE ON DUCTILITY OF STAINLESS STEEL DIN 1.4948. de Vries, M.I.; van der Schaaf, B.; Staal, H.U.; Elen, J.D. (Stichting Energieonderzoek Centrum Nederland, Petten). Oct 1978. 22p. (CONF-780722--25). Dep. NTIS (US Sales Only), PC A03/MF A01.

From 9. symposium on effects of radiation in structural materials; Richland, WA, USA (10 Jul 1978).

Test specimens of stainless steel DIN 1.4948, which is similar to AISI type 304, have been irradiated at 723 K and 823 K up to fluences of 1.10^{23} neutrons (n).m⁻² and 5.10^{24} n.m⁻² ($E > 0.1$ MeV). These are representative conditions for the SNR-300 reactor vessel and inner components after 16 years of operation. High temperature (723 K to 1023 K) tensile tests at strain rates (depsilon/dt) from 10^{-7} s⁻¹ to 10 s⁻¹ show a considerable decrease of tensile ductility. The extent depends on helium content, test temperature and strain rate. The atomic helium fractions of 3.10^{-7} and 7.10^{-6} result from the reactions of thermal neutrons with the 14 ppm boron, present in the steel. Helium embrittlement sets in at strain rates below 1 s⁻¹ to 10 s⁻¹ (the range of interest for Bethe-Tait accident analyses). A minimum total elongation value of 6% is shown at 923 K. The post-irradiation fatigue life is reduced by up to about 50% due to intergranular cracking. The combination of irradiation and fatigue causes a decrease of ductility after a smaller number of prior fatigue cycles than in the case of unirradiated material.

1054 (ECN--52) PSEUDO FISSION-PRODUCT CROSS SECTIONS FOR A 1300 MWE FAST BREEDER REACTOR (STATUS JULY 1978). Heijboer, R.J. (Stichting Energieonderzoek Centrum Nederland, Petten). Nov 1978. 26p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Zero-dimensional burn-up calculations have been performed for a reactor with an average core composition representative of a 1300 MWE LMFBR to obtain its fission product inventory at various time steps. In these calculations 162 fission-product nuclides have been taken into account. The fission yields used were taken from a preliminary version of ENDF/B-V. The adopted cross section set for the fission products was the adjusted RCN=2A group constant set for 37 important nuclides, supplemented with group constants calculated from other libraries, mainly from ENDF/B-IV. The adjustments in the capture cross sections are based upon integral data from STEK and CFRMF. In the calculations the complete fission-product mixture was separated into five mixtures of fission-products originating from the fissionable nuclides ²³⁵U, ²³⁸U, ²³⁹Pu, ²⁴⁰Pu and ²⁴¹Pu. For each of these heavy nuclides a 26-group set of pseudo fission-product cross sections was derived at each of the 9 time steps. Tables of the capture group constants of the 5 pseudo products are given at a burn-up stage of 41 MWd/kg. Comparisons are made with results of previous calculations (1976) in which other fission yields, capture cross sections and group weighting functions were used. On the average, the difference between the present and previous results is about +3%. The calculated capture cross sections of the pseudo products depend at present for about 82% on adjusted capture cross section data of individual fission-product nuclides (RCN=2A set). The uncertainty in the average pseudo fission-product capture cross section in a LMFBR, as calculated with the present data, is estimated at 6 to 9%.

1055 (ECN--59) FAST REACTOR PROGRAMME.

COMBINED SECOND AND THIRD QUARTERS 1978 PROGRESS REPORT. Hoekstra, E.K. (Stichting Energieonderzoek Centrum Nederland, Petten). Apr 1979. 62p. Dep. NTIS (US Sales Only), PC A04/MF A01.

Results of capture cross section adjustments of the Nd isotopes and ¹⁴⁷Pm are reviewed. Results of the in-pile canning failure experiment R54-F37 are compared with those of R54-F26 and R54-F42; progress is reported on the preparations being made for the HFR-TOP transient overpower experiments on irradiated fuel pins. Recently commenced thermo-chemical investigations on the intermetallic compounds URu₃, URh₃ and UPd₃ are discussed. The residual high-rate tensile ductility of stainless steel DIN 1.4948 measured after having applied various creep and fatigue damage fractions and results of tensile tests on post-low-cycle fatigue treated irradiated and non-irradiated DIN 1.4948 are given. Results of gas leak rate and aerosol penetration measurements on concrete specimens with artificially made cracks are reported on. Development of hydraulic computer codes, heat transfer measurements in the vicinity of grid-type spacers, planned Laser Doppler Anemometer measurements in a four-rod bundle and results of boiling experiments and temperature noise measurements behind a 34% blockage of a 28-rod bundle are discussed.

1056 (ECN--61) EFFECT OF NEUTRON IRRADIATION ON CREEP PROPERTIES OF 18CR-11NI (DIN 1.4948) STAINLESS STEEL PLATE AND WELDED JOINTS AT 823 K. van der Schaaf, B.; de Vries, M.I.; Elen, J.D. (Stichting Energieonderzoek Centrum Nederland, Petten). May 1979. 192p. Dep. NTIS (US Sales Only), PC A10/MF A01.

Design and safety analysis of permanent structures such as reactor vessel and shield tank of the loop type fast breeder reactor SNR-300 require post-irradiation creep data on the structural steel DIN 1.4948, which is equivalent to type AISI 304. Specimens from plate and weldments were irradiated at 823 K to thermal neutron fluences of about 5×10^{22} n.m⁻² and 2×10^{24} n.m⁻², (fast neutron fluences of 10^{23} n.m⁻² and 5×10^{24} ($E > 0.1$ MeV) respectively), followed by creep testing up to 10,000 h. The most important effect of irradiation is the diminished ductility of both plate and welded joints. As a result the creep strength of plate is reduced by 50 MPa which means a reduction in creep life by a factor of about 10. Welded joints fracture in unirradiated condition along the dendrite boundaries in the weld at a creep stress 20 MPa lower than parent metal. After irradiation the grain boundaries in the parent metal adjacent to the weld are weaker, causing a further reduction of creep strength of 30 MPa. The irradiation embrittlement is attributed to the formation of helium by thermal neutron reaction with the ¹⁰B isotope. The boron is mainly precipitated on the grain boundaries. In applying post irradiation creep-rupture data of type 304 ss for design and analysis of reactor components, thermal fluence limits rather than fast fluence limits should be set for life reduction factors and creep strains.

1057 (ECN--73) TENSILE PROPERTIES OF IRRADIATED AND FATIGUE EXPOSED STAINLESS STEEL DIN X 6 CRNI 1811 (SIMILAR TO AISI TYPE 304) PLATE AND WELDED JOINTS. de Vries, M.I.; van der Schaaf, B.; Elen, J.D. (Stichting Energieonderzoek Centrum Nederland, Petten). Oct 1979. 49p. Dep. NTIS (US Sales Only), PC A04/MF A01.

Test specimens of plate metal and welded

joints of stainless steel DIN 1.4948, which is similar to AISI type 304, have been irradiated at 723°K and 823°K up to fluences of 1.10^{23} n.m⁻² and 5.16^{24} n.m⁻² ($E > 0.1$ MeV). These are representative conditions for the SNR-300 reactor vessel and inner components after 16 years of operation. High-rate (depsilon/dt = 1 s^{-1}) tensile tests were performed after fatigue exposure up to various fractions of fatigue life (D) ranging from 5% to 95% at the same temperatures as the nominal temperatures of the irradiation series.

- 1058 (ECN--74) EFFECT OF NEUTRON IRRADIATION ON SEQUENTIAL CREEP-FATIGUE INTERACTION OF 18CR-11NI (DIN 1.4948) STAINLESS STEEL PLATE AND WELDED JOINTS AT 823 K. van der Schaaf, B.; de Vries, M.I.; Elen, J.D. (Stichting Energieonderzoek Centrum Nederland, Petten). Oct 1979. 162p. Dep. NTIS (US Sales Only), PC A06/MF A01.

The effect of irradiation on sequential creep-fatigue interaction was studied for plate and welded joints of 18Cr-11Ni stainless steel DIN 1.4948, equivalent to AISI type 304. Irradiation and test temperature were 823 K, the neutron fluence amounted to 10^{23} n.m⁻² ($E > 0.1$ MeV). Creep exposures lasted up to 4100 h. Low cycle fatigue testing was carried out with a total strain range of 0.6% and a strain rate of $3 \times 10^{-3} \text{ s}^{-1}$. The data are required for design and safety analysis of permanent structure such as: reactor vessel and shielded tank of the fast breeder reactor SNR-300.

- 1059 (EMD--76-13) EVALUATION OF THE STATUS OF THE FAST FLUX TEST FACILITY PROGRAM. (General Accounting Office, Washington, DC (USA)). 15 Nov 1976. 45p. General Accounting Office, Washington, DC.

Report to the Congress by the Comptroller General of the United States.

The report presents the current status of and problems associated with the construction and testing capabilities of the FFTF. Included are sections dealing with program cost experience, program schedule experience, and status of problems with major heat transport system components. (DG)

- 1060 (EMD--77-5) CONSIDERATIONS FOR COMMERCIALIZING THE LIQUID METAL FAST BREEDER REACTOR. (General Accounting Office, Washington, DC (USA)). 29 Nov 1976. 74p. General Accounting Office, Washington, DC.

Report to the Congress by the Comptroller General of the United States.

The report presents a picture of the commitment and effort necessary if the LMFBF and needed support technologies are ultimately to be commercialized. Emphasis is placed on the need for coordination of the planning and timing of the separate components required. Included are sections dealing with characteristics of a commercial LMFBF industry, schedule requirements for commercialization, alternative futures for commercialization of LMFBF technologies, constraints on commercialization, and conclusions and recommendations. (DG)

- 1061 (EMD--79-15) NUCLEAR REACTOR OPTIONS TO REDUCE THE RISK OF PROLIFERATION AND TO SUCCEED CURRENT LIGHT WATER REACTOR TECHNOLOGY. COMPTROLLER GENERAL'S REPORT TO THE CONGRESS. (General Accounting Office, Washington, DC (USA)). 23 May 1979. 54p. U.S. General Accounting Office, 441 G Street, N.W., Washington, DC.

In response to a request from the Vice

Chairman of the Joint Economic Committee and widespread interest by other Members of the Congress, GAO compared a number of alternative reactor technologies, a number of modifications to the reprocessing technology, and the thorium fuel cycle. GAO found that there is no proliferation-proof reactor technology, nor is there any technology which is clearly superior when compared in terms of resource utilization, licensability, environmental acceptability, cost, commercialization schedule, proliferation resistance, safeguardability, and economics. Therefore, GAO believes that Congress should not preclude the liquid metal fast breeder reactor option, at least at this time.

- 1062 (ENICO--1028) FAST REACTOR FISSION YIELDS FOR ²³³U AND ²³⁵U IRRADIATED IN ROW 4 OF EBR-II. Maeck, W.J.; Erikson, A.L.; Tromp, R.L. (Exxon Nuclear Idaho Co., Inc., Idaho Falls (USA)). Feb 1980. Contract AC07-79ID01675. 32p. Dep. NTIS, PC A03/MF A01.

Absolute ²³³U and ²³⁵U fast reactor fission yields are given for the stable and long-lived isotopes of krypton, rubidium, strontium, zirconium, molybdenum, xenon, cesium, barium, lanthanum, cerium, neodymium, and samarium, for samples irradiated in Row 4 of the Experimental Breeder Reactor II. The principal measurement technique for the fission product isotopes was isotope dilution mass spectrometry. The number of fissions was established by summing the number of fission product atoms in the heavy mass peak.

- 1063 (EPRI-AF--664(Vol.1)) COMPARATIVE STUDY AND EVALUATION OF ADVANCED-CYCLE SYSTEMS. FINAL REPORT. Pomeroy, B.D.; Fleck, J.J.; Marsh, W.D.; Brown, D.H.; Shah, R.P. (General Electric Co., Schenectady, NY (USA)). Feb 1978. 139p. Dep. NTIS, PC A07/MF A01.

A number of advanced energy-conversion concepts are now being proposed to supplement or supersede conventional power-generation technology. They are being proposed by individuals and organizations with widely varied backgrounds, using a diversity of approaches and assumptions for predicting performance, cost, and development requirements. The work reported here was undertaken to assist EPRI in planning R and D for the utility industry by analyzing 19 of the advanced concepts on a common basis using uniform technical and economic assumptions. The concepts range from a steam cycle with an atmospheric fluidized-bed furnace to longer-term options such as magnetohydrodynamic systems. The primary purpose of this study is to define techniques for assessing the worth of these concepts to the utility industry and the nation as a whole. Three methods have been developed: levelized cost of electricity; direct-weighting method; and net-present-worth method. These measure not only the life-cycle costs associated with each power plant concept, but also the intangible attributes such as development risk and reliability. They assess the relative importance of costs and intangibles in the context of utility goals.

- 1064 (EPRI-AF--664(Vol.2)(Pt.1)) COMPARATIVE STUDY AND EVALUATION OF ADVANCED-CYCLE SYSTEMS. FINAL REPORT. Pomeroy, B.D.; Fleck, J.J.; Marsh, W.D.; Brown, D.H.; Shah, R.P. (General Electric Co., Schenectady, NY (USA)). Feb 1978. 183p. Dep. NTIS, PC A09/MF A01.

This volume presents 3 appendices (A, B, and C) containing data dealing with the advanced power cycles evaluated. They are: Phase 1 Power Cycles Data Tabulation; Phase 2 Power Cycles--

Conceptual Designs; and Summary of Power Cycle Data and Development Plans from the Energy Conversion Alternatives Study. The 19 advanced cycles and their fuels evaluated in Phase 1 and two reference cycles (last two) are: advanced steam, atmospheric fluidized-bed furnace, coal; advanced steam, conventional furnace, No. 6 oil; advanced steam, high-temperature gas-cooled reactor, nuclear; advanced steam, liquid metal fast breeder reactor, nuclear; advanced open-cycle gas turbine, recuperative, air-cooled, high Btu gas derived from coal; advanced open-cycle gas turbine, recuperative, air-cooled, No. 6 oil; advanced open-cycle gas turbine, combined-cycle, air-cooled, low-Btu gas derived from coal; advanced open-cycle gas turbine, combined-cycle, water-cooled, low-Btu gas derived from coal; advanced open-cycle gas turbine combined-cycle, water-cooled, liquid semiclean fuel derived from coal; closed-cycle gas turbine, supercritical carbon dioxide, atmospheric fluidized-bed, coal; closed-cycle gas turbine, helium atmospheric fluidized-bed, coal; closed-cycle gas turbine, helium, high-temperature gas-cooled reactor, nuclear; open-cycle plasma MHD, coal; closed-cycle plasma MHD, conventional furnace, coal; liquid-metal MHD, atmospheric fluidized-bed, coal; metal-vapor turbine, atmospheric fluidized-bed, coal; thermionic, conventional furnace, coal; fuel-cell, low-temperature, hydrogen derived from coal; fuel-cell, low-temperature, No. 6 oil; conventional steam with stack gas scrubbing (reference case for base load and midrange), coal; and simple-cycle gas turbine (reference case for peaking), high-Btu gas derived from coal. (MCW)

- 1065 (EPRI-AF--664(Vol.2)(Pt.2)) COMPARATIVE STUDY AND EVALUATION OF ADVANCED-CYCLE SYSTEMS. FINAL REPORT. Pomeroy, B.D.; Fleck, J.J.; Marsh, W.D.; Brown, D.H.; Shah, R.P. (General Electric Co., Schenectady, NY (USA)). Feb 1978. 231p. Dep. NTIS, PC A11/MF A01.

Appendices D through H are included in this volume dealing with data compiled from the advanced cycle systems studies. They are: Details of Net Present Worth Approach; Net Present Worth With Discrete Annual Payments; Output of Net Present Worth Programs; Direct Weighting Method Power Cycle Ratings; and Direct Weighting Method: Utility Objectives and Impacts of Evaluation Criteria. (MCW)

- 1066 (EPRI-NP--645-SY(Vol.1)) ENGINEERING ASPECTS OF THE POOL TYPE LMFBR, 1000 MWE. FINAL REPORT. McDonald, J.S.; Golan, S. (Atoms International Div., Canoga Park, CA (USA); Bechtel Corp., San Francisco, CA (USA)). Apr 1978. 57p. Dep. NTIS (US Sale Only), MF A01.

The diverse directions of LMFBR development in the national programs here and abroad have established that both the loop-type and the pool-type of primary coolant systems should be considered to be viable options for an LMFBR power plant of practical size. Recognizing that the U.S. technological base for the pool-type plant was significantly smaller than that for the loop-type, this study of a 1000 MWe pool-type plant was concentrated on the areas of the plant unique to the pool concept to establish a more adequate basis for reassessment of the pool concept. The work concluded to date indicates that the pool concept is feasible with respect to design, construction and licensing practice in the United States. Volume 1 summarizes the work to date.

ASPECTS OF THE POOL TYPE LMFBR, 1000 MWE. FINAL REPORT. McDonald, J.S.; Golan, S. (Atoms International Div., Canoga Park, CA (USA); Bechtel Corp., San Francisco, CA (USA)). Apr 1978. 515p. Dep. NTIS (US Sale Only), MF A01.

Portions of document are illegible.

This volume presents the conceptual design and performance characteristics of a 1000-MWe pool-type Liquid Metal Fast Breeder Reactor (LMFBR) electric generating power plant.

- 1068 (EPRI-NP--645(Vol.2)(App.)) ENGINEERING ASPECTS OF THE POOL TYPE LMFBR, 1000 MWE. FINAL REPORT. McDonald, J.S.; Golan, S. (Atoms International Div., Canoga Park, CA (USA); Bechtel Corp., San Francisco, CA (USA)). Apr 1978. 262p. Dep. NTIS (US Sales Only), PC A12/MF A01.

Appendices to Volume 2 include detailed information under the following headings: NSSS system and equipment data sheets, deck trade study, rotatable plug and seal study, reflective insulation experience, fuel handling and storage, seismic design criteria, major component maintenance approach, radiation protection philosophy and criteria, NSSS design criteria, alternate to IHTS piping concepts, and safety licensing evaluation of IHTS and DDHRS containment isolation requirements.

- 1069 (EPRI-NP--645(Vol.3)) ENGINEERING ASPECTS OF THE POOL TYPE LMFBR, 1000 MWE. FINAL REPORT. Golan, S. (Bechtel Corp., San Francisco, CA (USA)). Apr 1978. 259p. Dep. NTIS (US Sale Only) MF A01.

Portions of document are illegible.

This volume reports the results of the balance-of-plant (BOP) work carried out under Phase A of the Prototype Large Breeder Reactor (PLBR) program sponsored by EPRI. The primary objective of BOP work for Phase A was to develop an integrated NSSS/BOP conceptual design under U.S. conditions for a 1000 MWe pool LMFBR utilizing a saturated steam cycle. The conceptual design was to be carried to sufficient depth to establish the technical feasibility of the major structures and systems unique to the pool concept and to investigate their practicality. This report describes the overall arrangement, major structures and systems, and provides engineering evaluations for key pool-related areas. Features affecting constructibility of a pool-type plant are also evaluated and areas requiring further study and resolution are identified. A detailed comparison between the pool and the loop was not performed; however, a number of general observations are noted.

- 1070 (EPRI-NP--646-SY) POOL-TYPE LMFBR PLANT, 1000 MWE PHASE A DESIGN. EXECUTIVE SUMMARY. Davies, S.M.; Golan, S.J. (General Electric Co., Sunnyvale, CA (USA); Bechtel Corp., San Francisco, CA (USA)). Apr 1978. 55p. Dep. NTIS, PC A04/MF A01.

The diverse directions of LMFBR development in the national programs here and abroad have established that both the loop-type and the pool-type of primary coolant systems should be considered to be viable options for an LMFBR power plant of practical size. Recognizing that the U.S. technological base for the pool-type plant was significantly smaller than that for the loop-type, this study of a 1000 MWe-type plant was concentrated on the areas of the plant unique to the pool concept to establish a more adequate basis for reassessment of the pool concept. The work concluded to date indicates that the pool concept is feasible with respect to design, construction and

- 1067 (EPRI-NP--645(Vol.2)) ENGINEERING

licensing practices in the United States.

- 1071 (EPRI-NP--646(Vol.1)) POOL-TYPE LMFBF PLANT 1000 MWE PHASE A DESIGN. PART II. HEAT TRANSPORT SYSTEMS, FINAL REPORT. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). Apr 1978. 215p. Dep. NTIS, PC A10/MF A01.

The results of a study on the design of heat transport systems for a 1000 MW(e) pool-type LMFBF are presented. The systems included decay heat removal as well as primary and intermediate heat transport systems.

- 1072 (EPRI-NP--646(Vol.2)) POOL-TYPE LMFBF PLANT 1000 MWE PHASE A DESIGN. PART II. APPENDIXES, FINAL REPORT. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). Apr 1978. 150p. Dep. NTIS, PC A07/MF A01.

A detailed study on the preliminary design of an intermediate heat exchanger for a 1000 MW(e) pool-type LMFBF is presented.

- 1073 (EPRI-NP--646(Vol.3)) POOL-TYPE LMFBF PLANT 1000 MWE PHASE A DESIGN. PART III. DECK STRUCTURE, FINAL REPORT. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). Apr 1978. 284p. Dep. NTIS, PC A13/MF A01.

The results of a detailed study on the design of the deck structure for a 1000 MW(e) pool-type LMFBF are presented.

- 1074 (EPRI-NP--646(Vol.4)) POOL-TYPE LMFBF PLANT 1000 MWE PHASE A DESIGN. PART IV. REACTOR ASSEMBLY AND CORE, FINAL REPORT. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). Apr 1978. 544p. Dep. NTIS, PC A23/MF A01.

The results of a detailed design study on the reactor assembly and core for a 1000 MW(e) pool-type LMFBF are presented.

- 1075 (EPRI-NP--646(Vol.5)) POOL-TYPE LMFBF PLANT 1000 MWE PHASE A DESIGN. PART IV. APPENDIX, FINAL REPORT. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). Apr 1978. 90p. Dep. NTIS, PC A05/MF A01.

The results of an engineering study of the design, fabrication, construction, and testing of the major structural components of a pool-type 1000 MW(e) LMFBF are presented.

- 1076 (EPRI-NP--646(Vol.6)) POOL-TYPE LMFBF PLANT 1000 MWE PHASE A DESIGN. PART V. AUXILIARY SYSTEMS. PART VI. RESEARCH AND DEVELOPMENT NEEDS, FINAL REPORT. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). Apr 1978. 239p. Dep. NTIS, PC A11/MF A01.

The results of an engineering study of auxiliary systems design and of additional research and development needs for a 1000 MW(e) pool-type LMFBF are presented.

- 1077 (EPRI-NP--646(Vol.7)) POOL-TYPE LMFBF PLANT 1000 MWE PHASE A DESIGN. PART VII. BALANCE-OF-PLANT FINAL REPORT. (Bechtel Corp., San Francisco, CA (USA)). Apr 1978. 455p. Dep. NTIS, PC A16/MFA01.

The results of an engineering study of the balance-of-plant design for a 1000 MW(e) pool-type LMFBF are presented.

- 1078 (EPRI-NP--646(Vol.8)) POOL-TYPE LMFBF PLANT 1000 MWE PHASE A DESIGN. PART VI. APPENDIXES, FINAL REPORT. (Bechtel Corp., San Francisco, CA (USA)). Apr 1978. 187p. Dep. NTIS, PC A09/MF A01.

Equipment lists, piping class sheets, plant drawings, and a Q-list (items subject to quality assurance program) for a 1000 MW(e) pool-type LMFBF are presented.

- 1079 (EPRI-NP--646(Vol.9)) POOL-TYPE LMFBF PLANT 1000 MWE PHASE A DESIGN. APPENDIX: INNOVATIVE DESIGN FEATURES, FINAL REPORT. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). Apr 1978. 42p. Dep. NTIS, PC A03/MF A01.

The results of a study of innovative design features for a 1000 MW(e) pool type LMFBF are presented.

- 1080 (EPRI-NP--846) SELF-ACTUATED SHUTDOWN SYSTEM FOR A COMMERCIAL SIZE LMFBF. FINAL REPORT. Dupen, C.F.G. (Combustion Engineering, Inc., Windsor, CT (USA)). Aug 1978. 155p. Dep. NTIS, PC A08/MF A01.

A Self-Actuated Shutdown System (SASS) is defined as a reactor shutdown system in which sensors, release mechanisms and neutron absorbers are contained entirely within the reactor core structure, where they respond inherently to abnormal local process conditions, by shutting down the reactor, independently of the plant protection system (PPS). It is argued that a SASS, having a response time similar to that of the PPS, would so reduce the already very low probability of a failure-to-scrum event that costly design features, derived from core disruptive accident analysis, could be eliminated. However, the thrust of the report is the feasibility and reliability of the in-core SASS hardware to achieve sufficiently rapid shutdown. A number of transient overpower and transient undercooling-responsive systems were investigated leading to the selection of a primary candidate and a backup concept. During a transient undercooling event, the recommended device is triggered by the associated rate of change of pressure, whereas the alternate concept responds to the reduction in core pressure drop and requires calibration and adjustment by the operators to accommodate changes in reactor power.

- 1081 (EPRI-NP--847) CCB, A TWO-DIMENSIONAL DIFFUSION THEORY CONTINUOUS CROSS-SECTION BURNUP CODE FOR FAST REACTOR ANALYSIS. FINAL REPORT. Borg, R.C.; Ott, K.O. (Purdue Univ., Lafayette, IN (USA)). School of Nuclear Engineering). Aug 1978. 219p. Dep. NTIS, PC A10/MF A01.

The burnup calculational procedures for fast breeder reactors are briefly reviewed; disadvantages of the procedures are discussed and several improvements are outlined and implemented. Microscopic group constants, which are usually considered time independent over each burnup interval, are replaced by time dependent expressions. These time dependencies are evaluated in terms of polynomials of the actinide number densities with predetermined coefficients. The burnup equations are then solved with time-dependent instead of the usual time-dependent one-group cross sections. Then, one-group flux calculations can replace most of the multigroup flux calculations for updating the power distribution. Use of these improvements reduces the computation time by more than a factor of two for a six-group problem, with very little inaccuracy in the predicted material inventory. A user's manual

for CCB is included. CCB is a two-dimensional, multigroup diffusion code for use in fast reactor burnup analysis. All group fluxes are in fast memory during the spatial calculation. The code includes techniques that have the primary goal of minimizing the number of outer iterations.

- 1082 (EPRI-NP--881-SY(Vol.1)) FURTHER EXAMINATION OF THE POOL TYPE 1000 MWE LMFBR. VOLUME 1. EXECUTIVE SUMMARY. McDonald, J.S.; Golan, S. (Atoms International Div., Canoga Park, CA (USA). Energy Systems Group; Bechtel National, Inc., San Francisco, CA (USA)). Aug 1978. 22p. Dep. NTIS, PC A02/MF A01.

A study of a 1000-MWe pool-type plant was initiated, concentrating on the areas of the plant unique to the pool concept. This four-volume interim report is the second presentation of results in this continuing study. Special emphasis is given to: (1) seismic requirements and related design impacts, (2) safety criteria and methodology to support U.S. licensing, (3) site fabrication and construction requirements and approaches, and (4) refinements in the design of the reactor tanks and supports, reactor cover, reactor internals, intermediate heat exchangers, pumps, head access area, and fuel handling system. The additional work supports the previous indication that the pool concept is feasible with respect to design, construction, and licensing practice in the United States.

- 1083 (EPRI-NP--881(Vol.2)) FURTHER EXAMINATION OF THE POOL TYPE 1000 MWE LMFBR. VOLUME 2. EXAMINATION OF NSSS CONCEPTS. INTERIM REPORT. (Atoms International Div., Canoga Park, CA (USA). Energy Systems Group). Aug 1978. 488p. Dep. NTIS, PC A21/MF A01.

The diverse directions of LMFBR development in the national programs, here and abroad, have established that both the loop-type and the pool-type of primary coolant systems should be considered to be viable options for an LMFBR power plant of practical size. Recognizing that the U.S. technological base for the pool-type plant was significantly smaller than that for the loop-type, a study of a 1000-MWe pool-type plant was initiated, concentrating on the areas of the plant unique to the pool concept. A four-volume interim report is the second presentation of results in this continuing study. Volume 2 deals with (1) nuclear steam supply system characteristics, (2) fabricability and constructibility, and (3) key pool-related design characteristics.

- 1084 (EPRI-NP--881(Vol.3)) FURTHER EXAMINATION OF THE POOL TYPE 1000 MWE LMFBR. VOLUME 3. EXAMINATION OF NSSS CONCEPTS (CONTINUED). INTERIM REPORT. (Atoms International Div., Canoga Park, CA (USA). Energy Systems Group). Aug 1978. 433p. Dep. NTIS, PC A19/MF A01.

The diverse directions of LMFBR development in the national programs, here and abroad, have established that both the loop-type and the pool-type of primary coolant systems should be considered to be viable options for an LMFBR power plant of practical size. Recognizing that the U.S. technological base for the pool-type plant was significantly smaller than that for the loop-type, a study of a 1000-MWe pool-type plant was initiated, concentrating on the areas of the plant unique to the pool concept. A four-volume interim report is the second presentation of results in this continuing study. Volume 3 deals with (1) plant performance; (2) safety and licensing; (3)

availability-maintainability, inspectability, reliability, and operability; and (4) development and test requirements.

- 1085 (EPRI-NP--881(Vol.4)) FURTHER EXAMINATION OF THE POOL TYPE 1000 MWE LMFBR. VOLUME 4. EXAMINATION OF BALANCE OF PLANT. INTERIM REPORT. (Bechtel National, Inc., San Francisco, CA (USA)). Aug 1978. 177p. Dep. NTIS, PC A09/MF A01.

The diverse directions of LMFBR development in the national programs, here and abroad, have established that both the loop-type and the pool-type of primary coolant systems should be considered to be viable options for an LMFBR power plant of practical size. Recognizing that the U.S. technological base for the pool-type plant was significantly smaller than that for the loop-type, a study of a 1000-MWe pool-type plant was initiated, concentrating on the areas of the plant unique to the pool concept. A four-volume interim report is the second presentation of results in this continuing study. Volume 4 contains the results of balance-of-plant (BOP) work performed by Bechtel under Phase A extension of the program. Although a conceptual BOP design for the pool was previously developed during Phase A, certain specific areas were selected for further study in order to more completely define design details and further develop the reference pool concept. A summary of the results of this work is presented. Five specific tasks were identified jointly with AI/CE. The scope of these tasks is described, and the task results are discussed. In addition, several modifications to the BOP arrangement were identified in order to improve structural design and constructibility aspects, and these are described. As a result of this work, the reference pool design drawings were updated and are contained in the appendix.

- 1086 (EPRI-NP--882-SY(Vol.1)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-1 DESIGN. PART I. EXECUTIVE SUMMARY. INTERIM REPORT. (General Electric Co., Sunnyvale, CA (USA); Bechtel National, Inc., San Francisco, CA (USA)). Sep 1978. 64p. Dep. NTIS, PC A04/MF A01.

A 900 MWe pool-type plant has been developed, to approximately the same level as current U.S. loop-type LMFBR plant designs, in a conceptual design study sponsored by EPRI. Concentrating on areas of the plant unique or especially important to the pool concept, this report covers the reactor deck; the design of internal structures, seismic response and construction of the reactor assembly; the thermal behavior of the plant during part-power operation and during transients; the auxiliary heat transport systems; the instrumentation and control systems; and the Balance of Plant. The main conclusion arising from this work is that the large pool-type LMFBR is a viable concept suitable for use in the United States.

- 1087 (EPRI-NP--882(Vol.2A)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-1 DESIGN. PART II. REACTOR ASSEMBLY: STRUCTURES. SECTIONS 1-6. INTERIM REPORT. Davies, S.M. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. 234p. Dep. NTIS, PC A11/MF A01.

The Phase A effort produced an initial description of the overall plant, structures and systems. During Phase A, General Electric developed the overall nuclear steam supply system (NSSS) conceptual design. It defined specific design approaches for selected NSSS components and subsystems after analyzing

various design alternatives. Bechtel assumed responsibility for the intermediate sodium piping arrangement, the access area above the reactor deck and the Balance of Plant (BOP). The resulting integrated plant design provided the necessary seismic data for both the NSSS and BOP.

- 1088 (EPRI-NP--882(Vol.2B)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-1 DESIGN. PART II. REACTOR ASSEMBLY: STRUCTURES. SECTIONS 7-11. INTERIM REPORT. Davies, S.M. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. 245p. Dep. NTIS, PC A11/MF A01.

The Phase A effort produced an initial description of the overall plant, structures and systems. During Phase A, General Electric developed the overall nuclear steam supply system (NSSS) conceptual design. It defined specific design approaches for selected NSSS components and subsystems after analyzing various design alternatives. Bechtel assumed responsibility for the intermediate sodium piping arrangement, the access area above the reactor deck and the Balance of Plant (BOP). The resulting integrated plant design provided the necessary seismic data for both the NSSS and BOP.

- 1089 (EPRI-NP--882(Vol.3)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-1 DESIGN. PART III: REACTOR ASSEMBLY - DECK. PART IV: REACTOR ASSEMBLY - FABRICATION. Davies, S.M. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. 254p. Dep. NTIS, PC A12/MF A01.

This report describes Phase A Extension 1 work performed between February 8 and July 31, 1978 on the design of a large pool-type LMFBR power plant. The work is the result of a team effort by Bechtel Corporation and General Electric Company which was sponsored and guided by the Electric Power Research Institute (EPRI). The objective of the work was to refine certain areas of design and bring them into better focus than had been provided by Phase A work performed between April 4 and December 30, 1977. Information is presented concerning the reactor assembly deck and fabrication of the reactor.

- 1090 (EPRI-NP--882(Vol.4)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-1 DESIGN. PART V. HEAT TRANSFER SYSTEM COMPONENTS. INTERIM REPORT. Davies, S.M. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. 445p. Dep. NTIS, PC A19/MF A01.

Portions of document are illegible.

The Phase A effort produced an initial description of the overall plant, structures and systems. During Phase A, General Electric developed the overall nuclear steam supply system (NSSS) conceptual design. It defined specific design approaches for selected NSSS components and subsystems after analyzing various design alternatives. Bechtel assumed responsibility for the intermediate sodium piping arrangement, the access area above the reactor deck and the Balance of Plant (BOP). The resulting integrated plant design provided the necessary seismic data for both the NSSS and BOP.

- 1091 (EPRI-NP--882(Vol.5)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-1 DESIGN. PART VI: REACTOR AUXILIARY SYSTEMS. PART VII: INSTRUMENTATION AND CONTROL. Davies, S.M. (General Electric Co., Sunnyvale, CA (USA).

Advanced Reactor Systems Dept.). Sep 1978. 249p. Dep. NTIS, PC A11/MF A01.

This report describes Phase A Extension 1 work performed between February 8 and July 31, 1978 on the design of a large pool-type LMFBR power plant. The work is the result of a team effort by Bechtel Corporation and General Electric Company which was sponsored and guided by the Electric Power Research Institute (EPRI). The objective of the work was to refine certain areas of design and bring them into better focus than had been provided by Phase A work performed between April 4 and December 30, 1977. Information is presented concerning auxiliary systems and instrumentation and control.

- 1092 (EPRI-NP--882(Vol.6)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-1 DESIGN. PART VIII: BALANCE OF PLANT. Golan, S. (Bechtel National, Inc., San Francisco, CA (USA)). Sep 1978. 328p. Dep. NTIS, PC A15/MF A01.

This report describes Phase A Extension 1 work performed between February 8 and July 31, 1978 on the design of a large pool-type LMFBR power plant. The work is the result of a team effort by Bechtel Corporation and General Electric Company which was sponsored and guided by the Electric Power Research Institute (EPRI). The objective of the work was to refine certain areas of design and bring them into better focus than had been provided by Phase A work performed between April 4 and December 30, 1977. Information is presented concerning the balance of plant systems.

- 1093 (EPRI-NP--883-SY) LARGE POOL LMFBR DESIGN. EXECUTIVE SUMMARY. Wett, J.F.; Churchill, J.R. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.; Stone and Webster Engineering Corp., Boston, MA (USA)). Aug 1978. 71p. Dep. NTIS, PC A04/MF A01.

The design effort reported and performed by ARD in the period February--July 1978, and by S and W from mid-March to July, 1978 concentrates on those parts of the nuclear island unique to a commercial size pool type LMFBR. In particular, the work covers the reactor vessel, deck, rotating plugs, upper and lower internals, internal plenum separator system, IHX, pumps, cold traps, intermediate system layout, containment/confinement system, plot plan, and residual heat removal systems. Preliminary thermal, hydraulic, stress and system analyses are also presented.

- 1094 (EPRI-NP--883(Vol.1)) LARGE POOL LMFBR DESIGN. FINAL REPORT. VOLUME 1. REACTOR. Wett, J.F. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Aug 1978. 193p. Dep. NTIS, PC A09/MF A01.

The design effort reported herein, and performed by ARD in the period February--July 1978, and by S and W from mid-March to July 1978, concentrates on those parts of the nuclear island unique to a commercial size pool type LMFBR. In particular, the work covers the reactor vessel, deck, rotating plugs, upper and lower internals, internal plenum separator system, IHX, pumps, cold traps, intermediate system layout, containment/confinement system, plot plan, and residual heat removal systems. Preliminary thermal, hydraulic, stress, and system analyses are also presented.

- 1095 (EPRI-NP--883(Vol.1)) LARGE POOL LMFBR DESIGN. VOLUME 1. REACTOR. FINAL REPORT. Wett, J.F. (Westinghouse Electric Corp.,

- Madison, PA (USA). Advanced Reactors Div.). Aug 1978. 195p. Dep. NTIS, PC A09/MF A01.
- The design effort concentrated on those parts of the nuclear island unique to a commercial size pool type LMFBR. In particular, the work covers the reactor vessel, deck, rotating plugs, upper and lower internals, internal plenum separator system, IHX, pumps, cold traps, intermediate system layout, containment/confinement system, plot plan, and residual heat removal systems. Preliminary thermal, hydraulic, stress and system analyses are also presented.
- 1096 (EPRI-NP--883(Vol.2)) LARGE POOL LMFBR DESIGN. FINAL REPORT. VOLUME 2. PLANT AND SYSTEMS. Wett, J.F. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Aug 1978. 76p. Dep. NTIS, PC A05/MF A01.
- The design effort reported and performed by ARD in the period February--July 1978, concentrates on those parts of the nuclear island unique to a commercial size pool type LMFBR. In particular, the work covers the reactor vessel, deck, rotating plugs, upper and lower internals, internal plenum separator system, IHX, pumps, cold traps, intermediate system layout, containment/confinement system, plot plan, and residual heat removal systems. Preliminary thermal, hydraulic, stress, and system analyses are also presented.
- 1097 (EPRI-NP--883(Vol.3)) LARGE POOL LMFBR DESIGN. VOLUME 3. APPENDIXES. FINAL REPORT. Wett, J.F. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Aug 1978. 493p. Dep. NTIS, PC A21/MF A01.
- Detailed studies are presented on: structural analysis, deck cooling analysis, vessel cooling and intermediate plenum analyses, radiation analysis, performance analysis, RHR system design, primary system flow and neutron flux measurements, alternate design considerations, selection of reference pump concept, intermediate heat exchanger design, passive RHR system, guidelines for EPRI LMFBR pool design, inert gas container for liquid metal containing pipe, insulation and support for LPR deck and rotatable plugs, selection of air as a deck coolant, two plena core inlets, reactor upper internals structure removal, inspection of in-vessel components, and upper internals/refueling chute interference.
- 1098 (EPRI-NP--883(Vol.4)) LARGE POOL LMFBR DESIGN. VOLUME 4. BALANCE OF PLANT. FINAL REPORT. (Stone and Webster Engineering Corp., Boston, MA (USA)). Aug 1978. 260p. Dep. NTIS, PC A12/MF A01.
- The primary objective of the balance-of-plant technical effort during Phase A Extension was to perform design work for a 1000 MWe (gross) pool type LMFBR to obtain an integrated plant design with emphasis of pool related structures and systems that differ from a loop type LMFBR or are of special concern. This report describes the general arrangement of major structures and equipment that are supported on a common foundation mat and the containment auxiliary systems that are unique or essential for the pool concept. Installation of major pool components during construction and containment design features that permit removal of major pool-related equipment are also described.
- 1099 (EPRI-NP--915) CRITIQUE OF THE PLBR
- PHASE A GUIDELINES. FINAL REPORT. Drechwa, Y.; Su, S.F.; Barthold, W.P. (Argonne National Lab., IL (USA)). Oct 1978. Contract W-31-109-ENG-38. 59p. Dep. NTIS, PC A04/MF A01.
- A critical analysis of the core/blanket complex as defined in the Phase A LMFBR Pool Guidelines was performed with respect to consistency and feasibility of the nuclear, thermal and mechanical design assumptions. Recommendations were made for changes in fuel assembly and blanket assembly design, core configuration and orificing.
- 1100 (EPRI-NP--948) RADIATION SOURCE TERMS IN LWR AND LMFBR SPENT FUEL. FINAL REPORT. Stamatelatos, M.G.; Huszar, L. (Science Applications, Inc., La Jolla, CA (USA)). Dec 1978. 149p. Dep. NTIS, PC A07/MF A01.
- The Electric Power Research Institute has sponsored an evaluation of the fission product gamma ray source terms for use in estimating radiation levels in the vicinity of spent nuclear fuel. This information is intended primarily for EPRI use in assessing how diversion-resistant LWR and LMFBR spent fuel will be during either long-term storage or various fuel reprocessing steps. Utility personnel engaged in the calculation of radiation levels around spent fuel pools and during refueling operations may find this report of use in their work.
- 1101 (EPRI-NP--1000(Vol.1)) OPTIMIZATION OF RADIALLY HETEROGENEOUS 1000-MW(E) LMFBR CORE CONFIGURATIONS. DESIGN AND PERFORMANCE OF REFERENCE CORES. RESEARCH PROJECT 620-25. Barthold, W.P.; Drechwa, Y.; Su, S.F.; Hutter, E.; Batch, R.V.; Beitel, J.C.; Turski, R.B.; Lam, P.S.K. (Argonne National Lab., IL (USA)). Nov 1979. Contract W-31-109-ENG-38. 164p. Dep. NTIS, PC A08/MF A01.
- A parameter study was conducted to determine the interrelated effects of: loosely or tightly coupled fuel regions separated by internal blanket assemblies, number of fuel regions, core height, number and arrangement of internal blanket subassemblies, number and size of fuel pins in a subassembly, etc. The effects of these parameters on sodium void reactivity, Doppler, incoherence, breeding gain, and thermohydraulics were of prime interest. Trends were established and ground work laid for optimization of a large, radially-heterogeneous, LMFBR core that will have low energetics in an HCDA and will have good thermal and breeding performance.
- 1102 (EPRI-NP--1000(Vol.2)) OPTIMIZATION OF RADIALLY HETEROGENEOUS 1000-MW(E) LMFBR CORE CONFIGURATIONS. APPENDIXES A AND B. RESEARCH PROJECT 620-25. Barthold, W.P.; Drechwa, Y.; Su, S.F.; Hutter, E.; Batch, R.V.; Beitel, J.C.; Turski, R.B.; Lam, P.S.K. (Argonne National Lab., IL (USA)). Nov 1979. Contract W-31-09-ENG-38. 82p. Dep. NTIS, PC A05/MF A01.
- A parameter study was conducted to determine the interrelated effects of: loosely or tightly coupled fuel regions separated by internal blanket assemblies, number of fuel regions, core height, number and arrangement of internal blanket subassemblies, number and size of fuel pins in a subassembly, etc. The effects of these parameters on sodium void reactivity, Doppler, incoherence, breeding gain, and thermohydraulics were of prime interest. Trends were established and ground work laid for optimization of a large, radially-heterogeneous, LMFBR core that will have low energetics in an HCDA and will have good

thermal and breeding performance.

- 1103 (EPRI-NP--1006(Vol.3)) OPTIMIZATION OF RADIALLY HETEROGENEOUS 1000-MW(E) LMFBR CORE CONFIGURATIONS. APPENDIX C. RESEARCH PROJECT 620-25. Barthold, W.P.; Drechwa, Y.; Su, S.F.; Hutter, E.; Batch, R.V.; Beitel, J.C.; Turski, R.B.; Lam, P.S.K. (Argonne National Lab., IL (USA)). Nov 1979. Contract W-31-109-ENG-38. 191p. Dep. NTIS, PC A09/MF A01.

A parameter study was conducted to determine the interrelated effects of: loosely or tightly coupled fuel regions separated by internal blanket assemblies, number of fuel regions, core height, number and arrangement of internal blanket subassemblies, number and size of fuel pins in a subassembly, etc. The effects of these parameters on sodium void reactivity, Doppler, incoherence, breeding gain, and thermohydraulics were of prime interest. Trends were established and ground work laid for optimization of a large, radially-heterogeneous, LMFBR core that will have low energetics in an HCDA and will have good thermal and breeding performance.

- 1104 (EPRI-NP--1000(Vol.4)) OPTIMIZATION OF RADIALLY HETEROGENEOUS 1000-MW(E) LMFBR CORE CONFIGURATIONS. APPENDIXES D AND E. RESEARCH PROJECT 620-25. Barthold, W.P.; Drechwa, Y.; Su, S.F.; Hutter, E.; Batch, R.V.; Beitel, J.C.; Turski, R.B.; Lam, P.S.K. (Argonne National Lab., IL (USA)). Nov 1979. Contract W-31-109-ENG-38. 83p. Dep. NTIS, PC A05/MF A01.

A parameter study was conducted to determine the interrelated effects of: loosely or tightly coupled fuel regions separated by internal blanket assemblies, number of fuel regions, core height, number and arrangement of internal blanket subassemblies, number and size of fuel pins in a subassembly, etc. the effects of these parameters on sodium void reactivity, Doppler, incoherence, breeding gain, and thermohydraulics were of prime interest. Trends were established and ground work laid for optimization of a large, radially-heterogeneous, LMFBR core that will have low energetics in an HCDA and will have good thermal and breeding performance.

- 1105 (EPRI-NP--1001) EPRI--CURL DYNAMIC ANALYSIS OF LOOP TYPE LMFBRs. FINAL REPORT. Cady, K.B.; Pavlenko, G.F.; Raber, J.R.; Khatib-Rahbar, M. (Cornell Univ., Ithaca, NY (USA)). May 1979. 516p. Dep. NTIS, PC A22/MF A01.

A computer-aided simulation is developed to analyze the transient behavior of loop type liquid metal-cooled fast breeder reactor plants. A set of nuclear, heat transfer, sodium dynamics, steam/water thermodynamics and control models for the multi-loop plant is developed to simulate operational, incident and accident transients. The major important features of the model include: (1) a multi-loop simulation, consisting of a single loop and the remaining lumped loops; (2) a method to calculate the characteristic times for every differential equation, and apply the Runge-Kutta algorithm to variables with slow and moderate time responses as compared to the integration time step and evaluate the variables with fast responses from a prompt approximation to assure numerical stability and reduce the computation time; and (3) a pre-accident initialization method by forcing most variables into their prompt approximation. The analytical submodels of the EPRI--CURL Code simulates: (a) the dynamics of sodium flows in the primary and secondary systems governed by forced and natural circulation; (b) reactor

heat transfer; (c) neutron kinetics, reactivity feedbacks, and reactor power control; (d) intermediate heat exchanger heat transfer; (e) sodium transport delays in the primary and secondary systems; (f) steam/water thermodynamics, normal feed water flow and auxiliary feed water flow control; and (g) turbine header, throttle, steam bypass and relief flow control. There is a total of 489 first order ordinary differential equations modelling the entire plant. The EPRI--CURL systems analysis computer code for loop-type liquid metal cooled fast breeder reactors is used to model the plant response of a typical LMFBR to several transient conditions.

- 1106 (EPRI-NP--1014-SY(Vol.1)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-2 DESIGN. PART I: EXECUTIVE SUMMARY. FINAL REPORT. Golan, S.; Davies, S.M. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.; Bechtel National, Inc., San Francisco, CA (USA)). Jun 1979. 49p. Dep. NTIS, PC A03/MF A01.

A 9000 MWe (net) pool-type LMFBR plant has been developed, to approximately the same level as current US loop-type LMFBR plant designs, in a conceptual design study sponsored by EPRI. A major guideline specified the plant to have an 875°F reactor outlet temperature and 550°F, 1000 psi steam with the design effort concentrating on areas of the plant unique or especially important to the pool concept. These areas include the reactor deck; the reactor assembly and its internal structures, the thermal behavior of the plant during steady state and transients; the auxiliary heat transport systems; the primary sodium pumps; the intermediate heat exchangers; the plant seismic response; maintenance and inspection; and the Balance of Plant. The main conclusion arising from this work is that the large pool-type LMFBR is a viable concept suitable for use in the United States.

- 1107 (EPRI-NP--1014(Vol.2)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-2 DESIGN. PART II: PLANT SUMMARY DESCRIPTION. FINAL REPORT. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jun 1979. 214p. Dep. NTIS, PC A16/MF A01.

This plant summary description volume covers the integrated Phase A plant conceptual design as it exists at the completion of Extension 2 work. In most cases the designs described are the reference arrived at after due consideration of reasonable alternatives. For a few specific design features, however, the lack of system and component development test data obscures the choice between alternatives. For these features certain alternate options remain under active interest. Examples of design features in this category are the upper internals structure, the pool thermal barriers and thermal insulation, the IHX flow distribution, and the hot pool thermal-hydraulic design in general.

- 1108 (EPRI-NP--1014(Vol.3)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-2 DESIGN. PART III: REACTOR ASSEMBLY, SEC. 1--2.5. FINAL REPORT. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jun 1979. 300p. Dep. NTIS, PC A13/MF A01.

The basic general arrangement of the reactor assembly for the General Electric Company's concept of a large pool type LMFBR plant was established in Phase A. EPRI design guidelines issued early in that phase specified fundamental plant features and requirements.

New arrangement drawings were prepared for Phase A Extension 1 and are contained in the Extension 1 report. The overall size and basic configuration of the reactor assembly is unchanged from Phase A. The Phase A Extension 2 work on the general arrangement is a progression from Extension 1. Prominent modifications are shown and described, and design uncertainties defined.

- 1109 (EPRI-NP--1014(Vol.4)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-2 DESIGN. PART III: REACTOR ASSEMBLY, SEC. 2-6--APPENDIX IIB. FINAL REPORT. (General Electric Co., Sunnyvale, CA (USA)). Advanced Reactor Systems Dept.). Jun 1979. 484p. Dep. NTIS, PC A21/MF A01.

The response to thermal striping and shock was determined for the upper internals structure. The components analyzed are identified and include (1) PRACS, (2) Fuel Transfer Tube, (3) Cold Trap, (4) Reactor Thermal Barrier, (5) Pump Thermal Barrier, (6) Horizontal Thermal Barrier, (7) IHX Upper Flow Shroud, (8) IHX Flow Shroud Inlet Region, (9) IHX External Cylinder, (10) Pump Inlet Nozzle, (11) Pump Exhaust Pipe, (12) Vessel Bottom Head, (13) Lower Core Support Structure, (14) Redundant Core Support, (15) Thermal Barrier Exits, and (16) Neutron Monitor Guide Tube.

- 1110 (EPRI-NP--1014(Vol.5)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-2 DESIGN. PART IV: REACTOR DECK. FINAL REPORT. (General Electric Co., Sunnyvale, CA (USA)). Advanced Reactor Systems Dept.). Jun 1979. 349p. Dep. NTIS, PC A15/MF A01.

The reactor roof or deck is a dominating structure in the pool reactor configuration; in addition to closing off the top of the reactor and providing access to the core for refueling (as it does also in the loop configuration) it also encloses and supports the entire primary coolant system including all major components such as pumps, IHXs, cold traps and instrumentation. Because of the absence of a suitable existing design basis in the United States for a large pool-concept design, the objective of the design effort was to establish feasibility through an intensive review of the many possible structural, insulation, cooling and shielding arrangements. The design effort considered construction, in-service inspection, response to non-routine conditions and influence on the pump and IHX configurations.

- 1111 (EPRI-NP--1014(Vol.6)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-2 DESIGN. PART V: HEAT TRANSPORT SYSTEMS. FINAL REPORT. (General Electric Co., Sunnyvale, CA (USA)). Advanced Reactor Systems Dept.). Jun 1979. 524p. Dep. NTIS, PC A22/MF A01.

The heat transport systems carry heat from the reactor to the IHXs and from the IHXs to the steam generators. The major components in the primary heat transport system are the primary sodium pumps and IHXs, and the major components in the intermediate heat transport system are the intermediate sodium pumps and steam generators. The history and evolution of the pool reactor heat transport systems and components during previous study phases are summarized. In addition, highlights of the current study are summarized.

- 1112 (EPRI-NP--1014(Vol.7)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-2 DESIGN. PART VI: AUXILIARY SYSTEMS; PART VII: PLANT CONTROL AND INSTRUMENTATION; PART VIII: SEISMIC ANALYSIS. FINAL REPORT. (General

Electric Co., Sunnyvale, CA (USA)). Advanced Reactor Systems Dept.). Jun 1979. 364p. Dep. NTIS, PC A16/MF A01.

Several key NSSS auxiliary systems are needed to support the reactor and main heat transport systems. These auxiliary systems include fuel handling and storage, reactor auxiliary cooling, sodium purification, and cover gas processing and control. Work has been performed on each of these systems, either to develop appropriate system conceptual designs or to investigate problem areas and identify solutions. For the fuel handling and storage system, the reactor auxiliary cooling system (RACS), and the primary sodium purification system overall system designs have been generated. Conceptual and/or preliminary designs of major components have been generated. For the cover gas system, two key aspects have been studied. The cover gas pressure control performance requirements have been defined and a suitable control system identified. In addition gas leakage through the component-to-deck seals has been analyzed. Recommended seal types and configurations have been established.

- 1113 (EPRI-NP--1014(Vol.8)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-2 DESIGN. PART IX: CONSTRUCTABILITY AND FABRICABILITY; PART X: MAINTAINABILITY AND INSPECTABILITY; PART XI: SAFETY. FINAL REPORT. (General Electric Co., Sunnyvale, CA (USA)). Advanced Reactor Systems Dept.). Jun 1979. 330p. Dep. NTIS, PC A15/MF A01.

Information is presented concerning the construction and fabrication of the upper internals structure and thermal barriers, maintainability and inspectability of reactor components, and design basis accidents.

- 1114 (EPRI-NP--1014(Vol.9)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-2 DESIGN. PART XII: BALANCE OF PLANT--PLANT DESCRIPTION. FINAL REPORT. (Bechtel, Inc., San Francisco, CA (USA)). Jun 1979. 262p. Dep. NTIS, PC A12/MF A01.

The Bechtel-developed BOP conceptual design, which combines the work of Phase A and its extensions, is described. The overall plant, structures, and systems of a 1000 MWe (gross) LMFBR utilizing the pool concept and the saturated steam cycle adopted in Phase II, are described. Particular attention is directed to the description of features unique to the pool concept, and to incorporating the results of the special studies done in Phase A and Extensions 1 and 2.

- 1115 (EPRI-NP--1014(Vol.10)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-2 DESIGN. PART XII: APPENDICES. FINAL REPORT. (Bechtel, Inc., San Francisco, CA (USA)). Jun 1979. 331p. Dep. NTIS, PC A15/MF A01.

Information presented includes system listing; an equipment list; piping class sheets; and a list of drawings.

- 1116 (EPRI-NP--1014(Vol.11)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION-2 DESIGN. PART XIII: BALANCE OF PLANT - EVALUATION OF POOL-RELATED AREAS. FINAL REPORT. (Bechtel, Inc., San Francisco, CA (USA)). Jun 1979. 525p. Dep. NTIS, PC A22/MF A01.

The supportive studies carried out in Phase A and its extensions investigated specific areas where practical solutions or procedures are considered essential prerequisites for subsequent design. Those areas identified for

each phase of the work are summarized and described in detail along with significant conclusions reached.

- 1117 (EPRI-NP--1014(Vol.12)) POOL-TYPE LMFBR PLANT 1000 MWE PHASE A-EXTENSION=2 DESIGN. FINAL REPORT. (Bechtel National, Inc., San Francisco, CA (USA)). Jun 1979. 109p. Dep. NTIS, PC A06/MF A01.

Drawings are presented concerning the reactor head compartment; equipment removal, reactor, and steam generator buildings; IHTS/IRACS piping enclosure; motor control layouts; maintenance and Na storage buildings; rectangular plant arrangement alternate 1; and general arrangement and reactor support buildings.

- 1118 (EPRI-NP--1015(SY)(Vol.1)) LMFBR POOL PLANT: 1000 MWE. VOLUME 1. EXECUTIVE SUMMARY. FINAL REPORT. McDonald, J.S.; Golan, S. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group; Bechtel National, Inc., San Francisco, CA (USA)). Feb 1979. 53p. Dep. NTIS, PC A04/MF A01.

Information is presented concerning reactor design features; steam supply system; safety; and reactor licensing.

- 1119 (EPRI-NP--1015(Vol.2)) LMFBR POOL PLANT: 1000 MWE. VOLUME 2. NSSS PLANT DESCRIPTION. FINAL REPORT. McDonald, J.S. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). Feb 1979. 416p. Dep. NTIS, PC A18/MF A01.

The report summarizes the design of the nuclear steam system; plant construction; plant performance; safety and licensing; and reactor availability.

- 1120 (EPRI-NP--1015(Vol.3)) LMFBR POOL PLANT: 1000 MWE. VOLUME 3. POOL REACTOR ENGINEERING, NSSS. FINAL REPORT. McDonald, J.S. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). Feb 1979. 401p. Dep. NTIS, PC A18/MF A01.

The report summarizes the safety criteria and methodology to support US licensing; plant performance; primary coolant circuit components; plant performance characteristics; and supporting test program.

- 1121 (EPRI-NP--1015(Vol.3)(App.)) LMFBR POOL PLANT: 1000 MWE. VOLUME 3. APPENDICES. FINAL REPORT. McDonald, J.S. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). Feb 1979. 337p. Dep. NTIS, PC A15/MF A01.

Appendices are presented concerning the structural design of reactor components, seismic effects, and operating procedures.

- 1122 (EPRI-NP--1015(Vol.4)) LMFBR POOL PLANT: 1000 MWE. VOLUME 4. BALANCE OF PLANT. FINAL REPORT. Golan, S. (Bechtel National, Inc., San Francisco, CA (USA)). Feb 1979. 177p. Dep. NTIS, PC A09/MF A01.

The report summarizes the refinements in the design of the reactor tanks and supports, reactor cover, reactor internals, intermediate heat exchangers, pumps, head access area, and fuel handling systems. The results of the work show that the pool concept is feasible with respect to design, construction, and licensing practice in the United States.

- 1123 (EPRI-NP--1016-SY) LARGE POOL LMFBR DESIGN: EXECUTIVE SUMMARY. Wett, J.F. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.; Stone and Webster Engineering Corp., Boston, MA (USA)). Mar 1979. 100p. Dep. NTIS, PC A05/MF A01.

The design effort reported is an extension of past design effort and continuous concentration on those parts of the nuclear island unique to a commercial size pool type LMFBR. In particular, the work covers the reactor vessel, deck, rotating plugs, upper and lower internals, internal plenum separator system, IHX, pumps, cold traps, intermediate system layout, containment/confinement system, plot plan, and residual heat removal systems. Preliminary thermal, hydraulic, stress and system analyses are also presented.

- 1124 (EPRI-NP--1016(Vol.1)) LARGE POOL LMFBR DESIGN. FINAL REPORT. Wett, J.F.; Churchill, J.R. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.; Stone and Webster Engineering Corp., Boston, MA (USA)). Mar 1979. 380p. Dep. NTIS, PC A17/MF A01.

The design effort reported is an extension on past design effort and continuous concentration on those parts of the nuclear island unique to a commercial size pool type LMFBR. In particular, the work covers the reactor vessel, deck, rotating plugs, upper and lower internals, internal plenum separator system, IHX, pumps, cold traps, intermediate system layout, containment/confinement system, plot plan, and residual heat removal systems. Preliminary thermal, hydraulic, stress, and system analyses are also presented.

- 1125 (EPRI-NP--1016(Vol.2)) LARGE POOL LMFBR DESIGN. VOLUME 2. BALANCE OF PLANT. FINAL REPORT. (Stone and Webster Engineering Corp., Boston, MA (USA)). Mar 1979. 282p. Dep. NTIS, PC A13/MF A01.

The work performed during this Phase A, Extension 2 is, in part, an extension of the conceptual design work for a 1000 Mwe (gross) large pool LMFBR initiated under Phase A, Extension 1 of the Phase A contract. This volume describes the general arrangement of the overall plant with emphasis on the nuclear island complex and facilities for maintenance of sodium heat transport system components. Auxiliary systems unique or essential for the pool concept were studied in more detail. A summary level construction schedule was developed for major activities of the nuclear island. The schedule and sequence for installing the large pool reactor primary heat transfer system components are presented in detail. Results are presented of the seismic evaluation of the nuclear island buildings and structural considerations that are of special concern to the pool type LMFBR.

- 1126 (EPRI-NP--1016(Vol.3)(Pt.1)) LARGE POOL LMFBR DESIGN. APPENDICES A-K. FINAL REPORT. Wett, J.F.; Churchill, J.R. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.; Stone and Webster Engineering Corp., Boston, MA (USA)). Mar 1979. 401p. Dep. NTIS, PC A18/MF A01.

Appendices are presented which describe component development, stress analysis, shielding calculations, and transient analysis.

- 1127 (EPRI-NP--1016(Vol.3)(Pt.2)) LARGE POOL LMFBR DESIGN. VOLUME 3, PART 2. APPENDICES L-CC. FINAL REPORT. Wett, J.F. (Westinghouse Electric Corp., Pittsburgh, PA

(USA). Advanced Reactors Div.; Stone and Webster Engineering Corp., Boston, MA (USA)). Mar 1979. 406p. Dep. NTIS, PC A18/MF A01.

These appendices contain information concerning design specifications for various reactor components and structures, reactor materials, and cost data.

- 1128 (EPRI-NP--1190-SR) EVAP USERS MANUAL. Heising, C.D.; James, J. Jr. (Massachusetts Inst. of Tech., Cambridge (USA)). Dept. of Nuclear Engineering; Electric Power Research Inst., Palo Alto, CA (USA)). Aug 1979. 47p. Dep. NTIS, PC A03/MF A01.

The computer code EVAP calculates the national economic value of an advanced electric power system - in particular, the liquid metal fast breeder reactor - under conditions of uncertainty in electricity demand and availability of technologies. On the basis of this calculation and assumptions about the effects of various reprocessing decisions on the availability of the advanced system, the code can also calculate the value of alternative reprocessing decisions. Uncertainties in the data are treated explicitly through the construction and evaluation of a Bayesian decision-analysis economic model (a decision tree). EVAP is short (400 lines of FORTRAN) and quick running (3 seconds of processing time on the Electric Power Research Institute's [EPRI's] PRIME-450 minicomputer).

- 1129 (EPRI-NP--1219) PRELIMINARY DESIGN: DUPLEX TUBE LOW-PRESSURE SATURATED STEAM GENERATOR FOR LARGE LMFBR PLANT. FINAL REPORT. Dawson, B.E. (Foster Wheeler Energy Corp., Livingston, NJ (USA)). Oct 1979. 253p. Dep. NTIS, PC A12/MF A01.

A preliminary design was completed for a steam generator which is applicable to a four loop LMFBR plant of 1000 MWe gross output employing dry and saturated steam conditions. Two steam generators of 364 Mwt thermal capacity would be in each loop. The steam generator is a straight, duplex tube design with a shell bellows for thermal expansion and a single tubesheet with an integral leak detection system. The report outlines a proposed series of development tasks to provide additional design data for the duplex tube and to demonstrate the steam generator design with test modules. The design can be adapted readily as either the evaporator or the superheater for a superheated steam cycle with temperatures up to at least 430°C (800°F), which is below the creep range for the 2 1/4 Cr-1 Mo material.

- 1130 (EPRI-NP--1265) ELECTROMAGNETIC PUMPS FOR LARGE POOL-CONCEPT LMFBR. Cage, J.F. Jr. (General Electric Co., Sunnyvale, CA (USA)). Dec 1979. 106p. Dep. NTIS, PC A06/MF A01.

A study was performed to evaluate application of Linear Induction Electromagnetic (EM) sodium pumps to the heat transport systems of a 1000 MW(e) pool-type sodium-cooled fast breeder reactor power plant. Characteristics of EM pumps were examined to evaluate their size and effect on plant efficiency, the arrangement of the primary heat transport system, and their potential advantage in mitigation of SCRAM temperature transients. Both gas-cooled pumps and pumps using high temperature insulation and cooled by the pumped sodium were studied. The parameters that establish size limitations were identified and several scoping arrangements designs were made. It was concluded that electromagnetic pumps are smaller and lighter than equivalent mechanical pumps, and that these features can be utilized to minimize the

size and complexity of the primary heat transport system.

- 1131 (EQL-Memo--8) URANIUM AVAILABILITY AND THE BREEDER DECISION. Holdren, J.P. (California Inst. of Tech., Pasadena (USA)). Environmental Quality Lab.). Jan 1974. 54p. California Inst. of Tech., Environmental Quality Lab., Pasadena.

The high priority assigned by the Federal government to the early development and commercial deployment of the Liquid Metal Fast Breeder Reactor (LMFBR) is attributed by some to the supposition that, without the breeder, a supply-price squeeze on uranium will soon materialize. This supposition is examined by considering the technology and economics of uranium utilization in nonbreeder reactors, in the context of available information about uranium resources at various prices and projections of the growth of nuclear power through 2020. Reactor characteristics, cost sensitivities, and estimates of uranium resources used are based largely on publications of the U.S. Atomic Energy Commission. The results show that existing reactor technologies--light-water reactors (LWRs), high temperature gas reactors (HTGRs), or a mix of these--could meet even the most enthusiastic projections of the expansion of nuclear generation through 2020 from presently known domestic uranium supplies, exploitable at \$50 per pound of U₃O₈ or less. The increment in electricity costs that arises from increasing uranium prices in the absence of commercial breeder reactors is about 1 mill/kwhe in 2000 and about 2 mills/kwhe in 2020 in the worst case (very high growth, no HTGRs), and significantly less in more plausible cases. Briefer attention is given to potential difficulties with rapid expansion of uranium mining operations, with enrichment capacity, and with environmental impact of mining low grade ores. Timely action in the first two areas would be required to meet high growth projections, but no fundamental obstacles are apparent. The environmental issue needs more study, but on present evidence does not constitute a persuasive case for an early commitment to the LMFBR. It is concluded that the urgency often ascribed to early deployment of LMFBRs on grounds of uranium availability is, in fact, illusory.

- 1132 (ERDA--77-56) LIFE: III FUEL-ELEMENT PERFORMANCE CODE. (Argonne National Lab., IL (USA)). 15 Jul 1977. Contract W-31-109-ENG-38. 96p. AT.

The LIFE-III computer code was developed to calculate the thermal and mechanical response of mixed-oxide fuel elements in a fast-reactor heat-transfer analysis and a finite-strain-theory structural analysis based on generalized plane strain and the method of successive elastic solutions. An incremental approach is used so that temperatures, stresses, and strains can be calculated during any specified history of reactor power cycling. Fuel-cladding and sodium-cladding chemical attack is treated by a cladding wastage model. Up to six axial sections are allowed to account for axial variations in power and coolant temperature. Fuel restructuring; migration of fabricated and swelling porosities; fuel and cladding thermal expansion, elasticity, swelling, and creep; fission-gas release; and fuel cracking, crack heating, and hot pressing are included in the analysis.

- 1133 (ESG-DOE--13239) SODIUM TECHNOLOGY AND COVER GAS SEAL DEVELOPMENT PROGRAMS.

QUARTERLY TECHNICAL PROGRESS REPORT, APRIL--JUNE 1978. Welch, F.H.; Steele, D.P. III; Hill, E.F. (Atomics International Div., Canoga Park, CA (USA)). 15 Aug 1978. Contract EY-76-C-03-0824-001. 30p. AT.

Progress is reported in the LMFBR seal test program and in sodium removal and decontamination of components.

1134 (ESG-DOE--13240) COVER GAS SEAL COMPONENTS. QUARTERLY TECHNICAL PROGRESS REPORT, APRIL--JUNE 1978. (Atomics International Div., Canoga Park, CA (USA). Energy Systems Group). 21 Aug 1978. Contract EY-76-C-03-0824. 23p. AT.

Progress is summarized in the program to develop and demonstrate the performance of commercial and custom-designed head region cavity seals for LMFBR reactors, to develop and evaluate advanced seal concepts capable of achieving improvement in radioactive gas containment and seal temperature-life capabilities, and to establish operational safety margins for head region seals for nominal and extraordinary LMFBR plant conditions. The CRBRP sodium dip seal configuration performance characteristics have been determined and the seal demonstrated to meet system requirements. Current activities include verifying the acceptability of CRBRP head region inflatable elastomer seal manufacturers, developing the technology to improve static inflatable seal design, and determining the ultimate temperature capabilities of several elastomer seal materials.

1135 (ESG-DOE--13241) HIGH-TEMPERATURE PIPING DESIGN TECHNOLOGY. QUARTERLY TECHNICAL PROGRESS REPORT, APRIL--JUNE 1978. (Atomics International Div., Canoga Park, CA (USA)). 30 Aug 1978. Contract EY-76-C-03-0824-026. 21p. AT.

The overall objectives of the High-Temperature Piping Design Technology progress is summarized in studies designed to: (1) provide design methodology for elevated-temperature piping systems, based on experiences with FFTF, STEC, and CRBRP designs and on operating high-temperature piping systems; (2) assess the validity of using current high-temperature Code Case 1592 design rules on piping systems; (3) investigate critical piping system design problems from the design criteria viewpoint; and (4) increase the confidence level in high-temperature design rules for piping by demonstrating the conservativeness of these rules.

1136 (ESG-DOE--13242) QUARTERLY TECHNICAL PROGRESS REPORT. REACTOR SAFETY, APRIL--JUNE 1978. (Rockwell International Corp., Canoga Park, CA (USA)). 30 Sep 1978. Contract EY-76-C-03-0824-028. 24p. Dep. NTIS, PC A02/MF A01.

Progress is summarized in LMFBR safety studies related to the characterization of sodium fires and fission products. Included are sections on high temperature-concentration aerosols, large-scale molten fuel tests, characterization of released sodium particles, and risk analysis. (DG)

1137 (ESG-DOE--13243) HIGH CYCLE FATIGUE OF 2-1/4 CR--1 MO STEEL AT TEMPERATURES TO 538°C (1000°F). Jaske, C.E.; Utah, D.A.; Frey, N.D. (Rockwell International Corp., Canoga Park, CA (USA)). 31 Jul 1978. Contract EY-76-C-03-0824. 43p. AT.

Long-life (greater than 10^5 cycles to

failure) fatigue experiments have been conducted on isothermally annealed 2-1/4 Cr--1 Mo steel in air at temperatures up to 538°C (1000°F). Strain was cycled continuously at rates equal to or greater than 4×10^{-3} sec⁻¹. Baseline experiments were for constant amplitude, fully reversed (zero mean stress) cycling. Exploratory experiments were conducted with tensile mean stress. Fatigue data for temperatures up to 427°C (800°F) were represented adequately by a single curve relating total strain range to fatigue life between 10^2 and 10^6 cycles to failure. A similar type of curve was developed for data at 538°C (1000°F). Tensile mean stress showed no reduction in fatigue resistance at 316°C (600°F), only a small reduction at 427°C (800°F), and a significant reduction at 538°C (1000°F). The degrading effect of mean stress at 427 and 538°C (800 and 1000°F) could not be accounted for by using a linear time-fraction plus cycle-fraction damage rule for creep-fatigue damage assessment. This result indicates that the effect of tensile mean stress must be taken into account by other means. It was found that both the Goodman approach and a stress-strain parameter method provided conservative and quite similar estimates of the effect of tensile mean stress on fatigue resistance. Two design fatigue curves from 10^2 to 10^6 cycles were developed for continuing cycling of 2-1/4 Cr--1 Mo steel in air and at strain rates equal to or greater than 4×10^{-3} sec⁻¹. One curve was for temperatures equal to or less than 427°C (800°F) and the other was for 538°C (1000°F). The maximum effect of mean stress was included in these curves using calculations based on the stress-strain parameter method.

1138 (ESG-DOE--13244) PRECONCEPTUAL STUDY OF 1000-MWE CARBIDE-FUELED LMFBR DESIGNS. Vitti, J.A.; Felten, L.D.; Brittingham, J.C.; Galluzzo, N.G.; Ostermier, B.J.; von Arx, A.V. (Atomics International Div., Canoga Park, CA (USA). Energy Systems Group). 30 Sep 1978. Contract EY-76-C-03-0824-018. 268p. AT.

A preconceptual study on the design of heterogeneous carbide-fueled LMFBR cores is summarized. The study was part of a broader effort on the design of proliferation-resistant LMFBR's. The study included (Pu-Th)c drivers with ThC blankets and (Pu-U)C drivers with UC or ThC blankets. It covered core configuration, nuclear analysis, thermal-hydraulic analysis, mechanical analysis, and economic evaluation.

1139 (ESG-DOE--13245) LMFBR COVER GAS SEALS DESIGN GUIDE ADDENDUM AND SEAL TECHNOLOGY SUMMARY. Dewart, N.A.; Welch, F.H.; Horton, P.H. (Atomics International Div., Canoga Park, CA (USA)). 30 Sep 1978. Contract EY-76-C-03-0824-011. 99p. AT.

Elastomer seals, metallic-elastomer seals, and a magnetic fluid seal were evaluated to LMFBR conditions. Basic permeation measurements and compression set resistance measurements were made. Some irradiation data were generated. Several new geometries and materials were disqualified for reactor use. In general, no new seal gave exemplary performance. Two metallic spring seal configurations with elastomeric jackets demonstrated potential worthiness under these test conditions, as did a magnetic fluid seal.

1140 (ESG-DOE--13246) ECEL THORIUM ASSEMBLY DESCRIPTIONS AND DATA SUMMARIES. Fiscella, J.M.; Specht, E.R. (Rockwell International Corp., Canoga Park, CA (USA)). 6 Oct 1978. Contract EY-76-C-03-0824-013. 250p. AT.

The Atomic International ECEL thorium-uranium-223 critical experiments covered a wide range of median fission energies (<0.1 eV to 0.4 MeV) and were used to measure a number of important reactor design parameters (fission ratios, reactivity worths, Doppler coefficients, etc.). The report describes ECEL assemblies 1 through 13 and summarizes the measured data that were obtained for the assemblies. The assemblies were constructed in pseudo-spherical configurations, with various combinations of thorium, uranium-233, uranium-238, moderator (carbon or beryllium) and aluminum (to mockup sodium coolant in the central test region.) This central test region was surrounded by a thorium decoupler and a thermal driver (uranium-235 and polyethylene.) Fission rate measurements (relative to uranium-235) were performed for uranium-233, -234, -236, and -238 as well as thorium-232, plutonium-239, and neptunium-237. Central reactivity worth measurements were made with 52 samples, including uranium-233, uranium-235, uranium-238, thorium-232, plutonium-239, sodium, aluminum, beryllium, carbon, boron, and stainless steel. Doppler measurements were made for uranium-235, uranium-238, thorium-232, and tantalum.

- 1141 (ESG-DOE--13247) SODIUM TECHNOLOGY PROGRAM QUARTERLY TECHNICAL PROGRESS REPORT, JULY--SEPTEMBER 1978. (Atomic International Div., Canoga Park, CA (USA)). 30 Oct 1978. Contract EY-76-C-03-0824-001. 17p. AT.
Progress is summarized in studies aimed at developing and evaluating techniques for sodium removal from LMFBR components and decontamination of LMFBR components.

- 1142 (ESG-DOE--13248) COVER GAS SEAL COMPONENTS. QUARTERLY TECHNICAL PROGRESS REPORT, JULY--SEPTEMBER 1978. (Atomic International Div., Canoga Park, CA (USA). Energy Systems Group). 10 Nov 1978. Contract EY-76-C-03-0824. 29p. AT.
Progress is summarized in the program to develop and demonstrate the performance of commercial and custom-designed head region cavity seals for LMFBR reactors, to develop and evaluate advanced seal concepts capable of achieving improvement in radioactive gas containment and seal temperature and life capabilities, and to establish operational safety margins for head region seals for nominal and extraordinary LMFBR plant conditions. The CRBRP sodium dip seal configuration performance characteristics have been determined, and the seal was demonstrated to meet system requirements. Current activities include verifying the acceptability of CRBRP head region inflatable elastomer seal manufacturers, developing the technology to improve static inflatable seal design, and determining the ultimate temperature capabilities of several elastomer seal materials. A new activity was initiated to design and analyze a flexing metal seal as an alternate concept to the inflatable elastomer seals.

- 1143 (ESG-DOE--13249) HIGH-TEMPERATURE PIPING DESIGN TECHNOLOGY. QUARTERLY TECHNICAL PROGRESS REPORT, JULY--SEPTEMBER 1978. (Rockwell International Corp., Canoga Park, CA (USA)). 10 Nov 1978. Contract EY-76-C-03-0824. 32p. TIC.

The overall objectives of the High-Temperature Piping Design Technology Program are: to provide design methodology for elevated-temperature piping systems, based on experiences with FFTF, ETEC, and CRBRP designs

and on operating high-temperature piping systems; to assess the validity of using current high-temperature Code Case 1592 design rules on piping systems; to investigate critical piping system design problems from the design criteria viewpoint; and to increase the confidence level in high-temperature design rules for piping by demonstrating the conservativeness of these rules.

- 1144 (ESG-DOE--13252) INTERMEDIATE-SIZE INDUCER PUMP PROGRAM. ANNUAL TECHNICAL PROGRESS REPORT, GOVERNMENT FISCAL YEAR 1978. (Atomic International Div., Canoga Park, CA (USA)). 26 Mar 1979. Contract EY-76-C-03-0824-020. 90p. AT.

The objective of the ISIP Development Program is to demonstrate the practical applicability of the Rockwell International inducer to large sodium pumps designed for reactor coolant service. The development of inducer pumps at Rockwell International started in the Rocketdyne Division with the development of high speed turbopumps for rocket engines. Since then inducer pumps have been applied successfully in the commercial area to water-jet propulsion systems for large high speed hydrofoil vessels. These commercial units require considerably greater life expectancy than those for space vehicles.

- 1145 (ESG-DOE--13254) ANNUAL TECHNICAL PROGRESS REPORT: REACTOR SAFETY, GOVERNMENT FISCAL YEAR 1978. (Atomic International Div., Canoga Park, CA (USA)). 31 Dec 1978. Contract EY-76-C-03-0824-028. 45p. Dep. NTIS, PC A03/MF A01.

Progress in LMFBR safety studies on accident debris behavior is reported under the following subtask titles: high-temperature-concentration aerosols, large-scale molten fuel tests, sodium release tests, and risk analysis.

- 1146 (ESG-DOE--13257) USE OF STRESS INDICES FOR PIPING UNDER ASME CODE CASE N-47. Rodabaugh, E.C. (Atomic International Div., Canoga Park, CA (USA)). 30 Sep 1978. Contract EY-76-C-03-0824. 83p. AT.

At the present time, rules for design of piping for nuclear power, elevated-temperature service are contained in the ASME Boiler Code Case N-47(-12). A Task Group under the Code Working Group on Piping Design (WGPD) has prepared a proposed addition on piping for a future revision of Code Case N-47. The WGPD proposed addition includes specific equations, using stress indices from NB-3600 of the Code, to implement the elastic analysis approach in Appendix T of Code Case N-47. The WGPD proposal was prepared when Code Case N-47(-7) was current; subsequent modifications to Code Case N-47 require modifications to the WGPD proposal; those modifications are shown in the report. A review of Appendix T of Code Case N-47 and the WGPD rules is presented, with the conclusion that the WGPD proposals are appropriate for use with the elastic analysis methods of the Code Case. Comparisons are made with the stress limits of ANSI B31.1, with the conclusion that for the particular conditions considered, Code Case N-47/WGPD limits are very conservative with respect to the limits of ANSI B31.1. Comparisons are also made with available test data. These comparisons also indicate that Code Case N-47/WGPD is conservative.

- 1147 (ESG-DOE--13261) SODIUM TECHNOLOGY PROGRAM. QUARTERLY TECHNICAL PROGRESS REPORT, OCTOBER--DECEMBER 1978. (Atomic International Div., Canoga Park, CA (USA)).

Energy Systems Group). 31 Jan 1979.

Contract EY-76-C-03-0824-001. 8p. AT.

Progress is the development of techniques for decontamination and sodium removal from LMFBR components is summarized.

- 1148 (ESG-DOE--13262) COVER GAS SEAL COMPONENTS AND REACTOR COMPONENTS: FUEL HANDLING. QUARTERLY TECHNICAL PROGRESS REPORT, OCTOBER--DECEMBER 1978. Steele, O.P. III; Horton, P.H. (Rockwell International Corp., Canoga Park, CA (USA)). 1 Feb 1979. Contract EY-76-C-03-0824-011. 15p. AT.

Progress is summarized in work to develop and demonstrate the performance of commercial and custom-designed head region cavity seals for LMFBR reactors, to develop and evaluate advanced seal concepts capable of achieving improvement in radioactive gas containment and seal temperature-life capabilities, and to establish operational safety margins for head region seals for nominal and extraordinary LMFBR plant conditions. During FY 1979, the major effort will be directed toward fabricating and performance testing two concepts for flexing metal dynamic seals as alternate concepts to inflatable elastomer seals.

- 1149 (ESG-DOE--13263) ACOUSTIC LEAK DETECTION-LOCATION. QUARTERLY TECHNICAL PROGRESS REPORT, OCTOBER--DECEMBER 1978. (Atomics International Div., Canoga Park, CA (USA). Energy Systems Group). 31 Jan 1979. Contract EY-76-C-03-0824-012. 8p. AT.

Progress in the development of acoustic emission techniques for detection of steam generator leaks in LMFBRs is summarized.

- 1150 (ESG-DOE--13264) REACTOR SAFETY. QUARTERLY TECHNICAL PROGRESS REPORT, OCTOBER--DECEMBER 1978. (Atomics International Div., Canoga Park, CA (USA)). 8 Jun 1979. Contract EY-76-C-03-0824. 44p. Dep. NTIS, PC A03/MF A01.

Progress is summarized in LMFBR safety studies related to the characterization of sodium fires and fission products. Included are sections on SOMIX Code development, sodium splash dispersal, aerosol leakage, aerosol model improvement, characterization of aerosol source term, large scale molten fuel tests, fuel and fission product release from burning sodium, and iodine attenuation.

- 1151 (ESG-DOE--13267) SIMPLIFIED DESIGN METHODS FOR BELLOWES. Jaquay, K.R.; Becht, C. IV. (Atomics International Div., Canoga Park, CA (USA). Energy Systems Group). 15 Feb 1979. Contract EY-76-C-03-0824-026. 227p. AT.

Over the past 3 years, the Energy Systems Group of Rockwell International has been working on a Department of Energy funded program to develop flexible piping joints for Class 1, nuclear piping systems. An important part of this program has been the design of gimbaled expansion joints to be used in a large-scale breeder reactor (LSBR) primary loop reactor outlet line. To support this activity, several design tools have been developed to simplify the analysis efforts. This report describes these design tools in sufficient detail to allow their use by other design organizations. Example problems are provided to clarify the procedures and to serve as check cases for method verification. The simplified design methods covered by this report are: (1) a program providing closed form stress calculations for angulated bellows (SCAB), (2) two finite element mesh generation programs for

U-shaped bellows convolutions (CONGEN/28 and CONGEN/15), and (3) elastic analysis-based bounding techniques for calculating creep ratchetting strains in bellows.

- 1152 (ESG-DOE--13268) RESEARCH AND DEVELOPMENT NEEDS TO DEVELOP DESIGN TOOLS FOR PREDICTION OF CREEP INSTABILITY OF FLEXIBLE PIPING JOINTS. (Rockwell International Corp., Canoga Park, CA (USA)). 15 Feb 1979. Contract EY-76-C-03-0824-019. 90p. AT.

A study was undertaken to define and plan a research program to develop design tools for the prediction of flexible piping joint (bellows) instability when operating in the creep regime. Currently, bellows instability is predicted by classical elastic beam column theory. This theory has sufficed for many years for the design of expansion joints for non-nuclear applications, however, such theory is not completely adequate for minimum risk design of expansion joints for 40 years service at elevated temperature. Design criteria that address instability with inelastic (creep or plastic) material behavior are necessary.

- 1153 (ESG-DOE--13271) STUDIES IN THE REDUCTION OF SODIUM CARRYOVER. Horton, P.H.; Oliva, R.M. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). 20 Jul 1979. Contract EY-76-C-03-0824. 71p. AT.

This study was undertaken to better define the mechanisms causing excessive carryover of sodium into the Clinch River Breeder Reactor Plant (CRBRP) - Ex-Vessel Transfer Machine (EVTM) chain hoist drive system. At the test temperature of 400°F, classical wetting of steels does not occur, but during development testing of the chain hoist drive system, apparent wetting was observed on the chain and other steel surfaces. Full scale and laboratory scale tests were used to study the wetting mechanism with emphasis on the influence of oxygen and moisture in the argon cover gas atmosphere. It was demonstrated that moisture enrichment of the atmosphere (tests were conducted at 50 ppm moisture) encourages adherence of sodium to the metal surfaces while oxygen enrichment showed no such effect.

- 1154 (ESG-DOE--13272) SODIUM TECHNOLOGY PROGRAM. QUARTERLY TECHNICAL PROGRESS REPORT, JANUARY--MARCH 1979. (Atomics International Div., Richland, WA (USA). Rockwell Hanford Operations). 31 May 1979. Contract EY-76-C-03-0824-027. 18p. AT.

Progress in sodium evaporation process development and decontamination studies applicable to the FFTF, CRBRP, and LMFBR target plant programs is reported.

- 1155 (ESG-DOE--13273) COVER GAS SEAL COMPONENTS AND REACTOR COMPONENTS: FUEL HANDLING. QUARTERLY TECHNICAL PROGRESS REPORT, JANUARY--MARCH 1979. (Atomics International Div., Canoga Park, CA (USA)). 15 Jun 1979. Contract EY-76-C-03-0824-011. 28p. AT.

The objectives of the Cover Gas Seal Component test program are to develop and demonstrate the performance of commercial and custom-design head region cavity seals for LMFBR reactors, to develop and evaluate advanced seal concepts capable of achieving improvement in radioactive gas containment and seal temperature-life capabilities, and to establish operational safety margins for head region seals for nominal and extraordinary LMFBR plant conditions. During FY 1979, the

major effort will be directed toward fabricating and performance testing two concepts for flexing metal dynamic seals as alternate concepts to inflatable elastomer seals. The objectives of the Fuel Handling Subcomponent development program are to verify that wetting of the chain will reduce sodium carryover into the EVTM drive components and determine whether or not the chain will remain wetted throughout the EVTM life cycle.

- 1156 (ESG-DOE--13274) HIGH-TEMPERATURE PIPING DESIGN TECHNOLOGY. SEMI-ANNUAL TECHNICAL PROGRESS REPORT, OCTOBER 1978--MARCH 1979. (Atomics International Div., Richland, WA (USA). Rockwell Hanford Operations). 1 Jun 1979. Contract EY-76-C-03-0824-026. 18p. AT.

Progress is reported in the following LMFBR piping design subtasks: (1) evaluation of sensitivity of piping design margins to variability in materials and geometry, (2) high-cycle fatigue at elevated temperature, and (3) piping component tests.

- 1157 (ESG-DOE--13275) ACOUSTIC LEAK DETECTION-LOCATION. QUARTERLY TECHNICAL PROGRESS REPORT, JANUARY--MARCH 1979. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). 20 Jun 1979. Contract EY-76-C-03-0824-012. 8p. AT.

The objective of this program is to develop a multidimensional detection-location system for steam generator tube leaks. Verification of the detection method will be made by injecting pressurized gas into water in the Hydraulic Test Model at General Electric. Typical background patterns from an operating steam generator will be monitored and recorded using a steam generator at EBR-II. Using data from these and other sources, hard- and software will be developed which can be used for the detection-location of steam generator leaks.

- 1158 (ESG-DOE--13278) SODIUM TECHNOLOGY PROGRAM. QUARTERLY TECHNICAL PROGRESS REPORT, APRIL-JUNE 1979. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). 24 Aug 1979. Contract EY-76-C-03-0824-001. 13p. AT.

Progress is reported for the following LMFBR-CRBR related activities: (1) sodium removal process development, and (2) decontamination studies.

- 1159 (ESG-DOE--13279) COVER GAS SEAL COMPONENTS. QUARTERLY TECHNICAL PROGRESS REPORT, APRIL--JUNE 1979. (Rockwell International Corp., Canoga Park, CA (USA)). 24 Aug 1979. Contract EY-76-C-03-0824-011. 13p. AT.

Progress is summarized in efforts to develop and demonstrate the performance of commercial and custom-designed head region cavity seals for LMFBR reactors, to develop and evaluate advanced seal concepts capable of achieving improvement in radioactive gas containment and seal temperature-life capabilities, and to establish operational safety margins for head region seals for nominal and extraordinary LMFBR plant conditions. During FY 1979, the major effort has been directed toward designing and analyzing concepts for flexing metal dynamic seals as alternate concepts to inflatable elastomer seals. In June, the program was redirected to concentrate on improving and simplifying fabrication techniques and designs of inflatable elastomer seals.

- 1160 (ESG-DOE--13280) ACOUSTIC LEAK DETECTION-LOCATION. QUARTERLY TECHNICAL PROGRESS REPORT, APRIL-JUNE 1979. Thiele, A.W. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). 31 Aug 1979. Contract EY-76-C-03-0824-012. 20p. AT.

Research in developing a multidimensional location system for steam generator tube leaks is reported. Measurements of the acoustic background were made at EBR-2 during flow calibration. The background signals were proportional to flow, but were not dependent upon power level. (FS)

- 1161 (ESG-DOE--13281) REACTOR SAFETY. QUARTERLY TECHNICAL PROGRESS REPORT, JANUARY-MARCH 1979. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). 28 Sep 1979. Contract AT03-76SF74028. 32p. Dep. NTIS, PC A03/MF A01.

Progress in LMFBR safety studies is briefly summarized.

- 1162 (ESG-DOE--13284) REACTOR SAFETY. QUARTERLY TECHNICAL PROGRESS REPORT, APRIL-JUNE 1979. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). 15 Oct 1979. Contract W-31-109-ENG-38. 25p. Dep. NTIS, PC A02/MF A01.

Progress in LMFBR safety studies is briefly summarized.

- 1163 (ESG-DOE--13285) CRBRP SODIUM FIRE TESTS. TECHNICAL PROGRESS REPORT, MAY-JUNE 1979. (Rockwell International Corp., Richland, WA (USA). Rockwell Hanford Operations). 15 Oct 1979. Contract W-31-109-ENG-38. 11p. Dep. NTIS, PC A02/MF A01.

Progress in small-scale fire suppression tests and spray maximization tests related to sodium fires in the Clinch River Breeder Reactor is reported.

- 1164 (ESG-DOE--13290) SODIUM TECHNOLOGY PROGRAM. QUARTERLY TECHNICAL PROGRESS REPORT, JULY-SEPTEMBER 1979. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). 24 Oct 1979. Contract A103-76SF70027. 19p. AT.

Progress is reported for the following LMFBR-CRBR related activities: (1) sodium evaporation process development and (2) decontamination studies.

- 1165 (ESG-DOE--13294) REACTOR SAFETY. ANNUAL TECHNICAL PROGRESS REPORT, GOVERNMENT FISCAL YEAR 1979. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). 15 Apr 1980. Contract AT03-76SF74028. 46p. Dep. NTIS, PC A03/MF A01.

Information is presented on LMFBR reactor safety concerning the energetics effects of sodium spray fires; sodium drop and spray burning; core debris accommodation; attenuation in containment; and attenuation in the environment.

- 1166 (ESG-DOE--13310) FAST REACTOR FLUENCE DOSIMETRY. TECHNICAL PROGRESS REPORT, OCTOBER 1978-MARCH 1979. Oliver, B.M.; Bradley, J.G.; Farrar, H. IV. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). 15 Feb 1980. Contract AT03-76SF75028. 64p. Dep. NTIS, PC A04/MF A01.

Information is presented concerning the development and performance testing of helium

accumulation fluence monitors and the measurement of helium distribution in irradiated unfueled Type 316 stainless steel fuel cans.

- 1167 (ESG-DOE--13312) CRBRP SODIUM FIRE TESTS. QUARTERLY TECHNICAL PROGRESS REPORT, OCTOBER-DECEMBER 1979. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). 29 Feb 1980. Contract AM03-76SF00824. 13p. Dep. NTIS, PC A02/MF A01.

The objective of the small-scale group of tests is to demonstrate that sodium will drain from the surface of the fire suppression deck into the catch pan without interference during a spill event, and to demonstrate that burning is terminated following the spill event by the accumulation of combustion products inside the drain pipes. The results of this series of tests will be used to validate the catch pan fire suppression deck design concept based on the criteria that sodium will drain freely from the surface of the fire suppression deck and that burning is terminated in an acceptably short time (less than or equal to 36 h). The objective of the large-scale group of tests is to provide experimental data on the consequences of sodium spills using prototypic leak rates and to demonstrate the effectiveness of a large-scale fire suppression Q-deck assembly.

- 1168 (ESG-DOE--13313) FAST REACTOR FLUENCE DOSIMETRY. TECHNICAL PROGRESS REPORT, APRIL-SEPTEMBER 1979. Bradley, J.G.; Oliver, B.M.; Farrar, H. IV. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). 31 Mar 1980. Contract AT03-76SF74028. 51p. Dep. NTIS, PC A04/MF A01.

Information is presented concerning the development and performance testing of helium accumulation fluence monitors.

- 1169 (ETEC--78-9) SODIUM--WATER REACTION PRODUCT DEPOSITION AND POST-TEST CLEANING OF THE MODULAR STEAM GENERATOR. Neely, H.H.; McCollum, W.A.; Shepard, R.C.; Stearns, J.D. (Energy Technology Engineering Center, Canoga Park, CA (USA)). 1 Dec 1978. Contract EY-76-C-03-0700. 50p. AT.

Six sodium--water reaction tests have been completed in the Atomics International Modular Steam Generator at prototypic LMFBR plant conditions. Tests were run in preprogrammed regions of the generator and resulted in reaction product deposition that was appreciably different for each of the three test locations. Deposition is shown to be unique to test location. Major deposition that can impede sodium flow in a generator is generated from a leak above the lower relief line, which results in a pool reaction. After testing, the steam generator was satisfactorily cleaned with Dowanol for post-test destructive examination. Dowanol and sodium--water reaction products were shown to generate water during the cleaning process, which, in turn, appreciably enhanced the reaction rate of available metallic sodium.

- 1170 (ETEC--78-10) SODIUM--WATER REACTION TEST IN LLTR. SERIES I. FINAL REPORT. Eichelberger, R.L. (Energy Technology Engineering Center, Canoga Park, CA (USA)). 15 Jul 1978. Contract EY-76-C-03-0700. 247p. TIC.

Portions of document are illegible. Extensive large-leak testing is being conducted to investigate effects of sodium-

water reaction events in LMFBR steam generators. The general objectives of the program are to provide data for validating analytical design methods, to provide data on secondary failures, and to provide performance data on ancillary and cleanup systems. The first test series, utilizing the Atomics International Modular Steam Generator as the test article, has been successfully completed and is reported. For this series, two double-ended guillotine (DEG) tests were conducted in the vertical section of the steam generator and four DEG tests were conducted in the horizontal (hockey-stick) region. The first five tests were DEG ruptures of a standard steam generator tube. The sixth test rupture tube had a flow area equivalent to three standard tubes. Results from these six tests have indicated that analytical methods are conservative (i.e., predict higher reaction temperatures and pressures than measured). Salient test results are presented as well as comparisons between the six tests performed. The inspection activity was successful in identifying the consequences of a major sodium-water reaction event. Among the notable observations were: secondary tube wastage, secondary tube bowing, reaction product deposition, and rupture relief characteristics.

- 1171 (ETEC--78-11) METALLURGICAL EVALUATION OF AN FFTF PPS PRESSURE TRANSDUCER INTERFACE SEAL ASSEMBLY AFTER LIQUID SODIUM SERVICE. Thompson, E.G. (Rockwell International Corp., Canoga Park, CA (USA)). 30 Sep 1978. Contract EY-76-C-03-0700. 39p. AT.

The interface seal assembly of an RDT C 6-1T pressure transducer system, Model S 63-4, Serial No. 1, manufactured by Statham Instruments (now Gould, Inc. Measurement System Division) was destructively examined after approximately 23,000 h of liquid sodium service. The purpose of examination was to determine if service degradation had taken place in this bellows-type assembly.

- 1172 (ETEC--78-12) METALLURGICAL EVALUATION OF THE MODULAR STEAM GENERATOR (MSG) AFTER LLTR TESTING. Stearns, J.D. (Energy Technology Engineering Center, Canoga Park, CA (USA)). 30 Sep 1978. Contract EY-76-C-03-0700. 49p. AT.

The AI LMFBR modular steam generator was subjected to a thorough metallurgical examination during final disassembly. No major anomalies were found. Tube deformation and localized wastage were found near the programmed leak sites. No wastage greater than 0.048 cm (0.019 in.) was found, and at most locations was much less. Less than 10 kg (22 lb) of residual metallic sodium and sodium reaction products were found. The residual sodium was found at locations where flow was restricted during organic solvent circulation.

- 1173 (ETEC--78-15) FUEL CLADDING: WIRE WRAP WEAR IN SODIUM. Droher, J.J.; Hoffman, N.J.; Stearns, J.D. (Energy Technology Engineering Center, Canoga Park, CA (USA)). 30 Sep 1978. Contract EY-76-C-03-0700. 59p. Dep. NTIS, PC A04/MF A01.

Tests involving longitudinal-motion and transverse-motion rubbing of fuel cladding by wire wrap were performed in ambient air and in elevated temperature sodium to determine the type and magnitude of wear of the fuel cladding. Adhesive wear was encountered in tests in sodium at 900° to 1300°F (480 to 705°C), while abrasive wear was encountered in tests in sodium at 450°F (205°C) and in ambient air. The wear scar generally did not exceed 2

or 3 mils (0.05 or 0.08 mm). In all of the tests, there was not a high degree of correlation of depth of wear scar with such test parameters as sodium temperature, contact load, wire wrap pitch or angle, stroke length, rubbing cycle, or mode of rubbing. In addition to the metal removed, a layer of disturbed metal was found under some of the wear scars.

- 1174 (ETEC--78-16) SODIUM AEROSOL AND HYDROGEN GENERATION TESTS. Davis, K.A. (Energy Technology Engineering Center, Canoga Park, CA (USA)). 15 Nov 1978. Contract EY-76-C-03-0700. 245p. AT.

Results are reported on a testing program designed to evaluate the factors which influence sodium aerosol and hydrogen generation from a sodium leak. The factors identified for evaluation included leak rate, temperature, moisture and oxygen concentration, and leak surface geometry and orientation. The range of parameters was selected to provide data pertinent to leak detection in the FFTF inerted cells. Fifty-four tests were performed. Details of the test method and results of each test are presented. Photographs are presented to show the effect of test parameter variation on the behavior of the leaking sodium. Graphs showing the effects of each parameter on aerosol and hydrogen generation are presented.

- 1175 (ETEC--78-18) SODIUM IONIZATION DETECTOR RESPONSE TO EXTRANEOUS MATERIALS. Davis, K.A.; Klee, J.A.; Douglas, R.E. (Energy Technology Engineering Center, Canoga Park, CA (USA)). 31 Oct 1978. Contract EY-76-C-03-0700. 70p. AT.

Results are reported on a series of tests to determine what materials commonly existing in a nuclear reactor sodium heat transport system may form an aerosol to which a Westinghouse sodium ionization detector may respond. Thirty-five materials were tested. The results showed a relatively high response to several of these indicating the possibility that a false indication of a sodium leak could be presented. A ranking of the materials is presented to show the relative sensitivity of the detector to these materials.

- 1176 (ETEC--79-1) ENERGY TECHNOLOGY ENGINEERING CENTER (ETEC). ANNUAL TECHNICAL PROGRESS REPORT, JANUARY--DECEMBER 1978. (Energy Technology Engineering Center, Canoga Park, CA (USA)). 30 Apr 1979. Contract EY-76-C-03-0700. 382p. AT.

Progress is reported for the following LMFBR-related activities: sodium-water reaction testing, subscale inducer pump test, hydraulically supported absorber ball shutdown assembly, articulated control assembly, pipe clamp test program, CRBR small sodium valve testing, gate valve testing, sodium flowmeter testing, pressure transducer testing, sodium-to-gas leak detection, sodium level measurement system testing, electrical resistance heater tests, proximity transducers, strain measurement, development of percussion welding applications, and friction and wear in sodium. (JWR)

- 1177 (ETEC-TDR--78-13) ALIP/FLOWMETER CALIBRATION. De Vita, V. (Energy Technology Engineering Center, Canoga Park, CA (USA)). 8 Sep 1978. Contract EY-76-C-03-0700. 58p. AT.

This report describes the calibration of an Annular Linear Induction Pump (ALIP), a one-inch electromagnetic (EM) flowmeter, and a one-inch permanent magnet (PM) flowmeter. The three

items, incorporated into one test article, were calibrated at 600, 700, and 800°F (315.5, 371.1, 426.7°C) over a liquid sodium flow range from 1 to 10 gpm (6.3 to 63 x 10⁻⁵m³/sec) and for pump differential pressures from 5 to 25 psi (34.5 to 172.5 kPa). The test articles were supplied by HEDL, Richland, Washington in support of the Mark II=C Loop Test Program which is part of the National LMFBR Mixed Oxide Fuel Transient Performance program. The reference flow for this calibration was determined by forcing sodium from one supply tank to a receiver tank. By measuring the change in the sodium free surface level in the supply tank as a function of time, the volumetric flow rate was computed.

- 1178 (ETEC-TDR--79-4) ENDURANCE TESTING OF PROTOTYPIC SUBSCALE INDUCER PUMP AT SCTL. Baker, R.S. (Energy Technology Engineering Center, Canoga Park, CA (USA)). 26 Jul 1979. Contract EY-76-C-03-0700. 45p. AT.

A series of tests have been conducted in sodium on a prototypic subscale inducer pump designed by Atomics International. Pump operation has been successful. Testing consisted of three intervals totaling 2000-h.

- 1179 (ETEC-TDR--79-6) EVALUATION OF SPTF LIQUID RHEOSTAT FOR ISIP TEST PROGRAM. Baker, R.S. (Energy Technology Engineering Center, Canoga Park, CA (USA)). 22 Aug 1979. Contract EY-76-C-03-0700. 8p. AT.

This report is an evaluation of the performance of the liquid rheostat installed at SPTF for sodium pump speed control.

- 1180 (ETEC-TDR--79-7) STRESSCOAT TEST OF 6-INCH PIPE CLAMP IN TTF: ARTICLE B. Rollins, C.T. (Energy Technology Engineering Center, Canoga Park, CA (USA)). 4 May 1979. Contract EY-76-C-03-0700. 49p. AT.

The Stresscoat evaluation is presented for the stress distribution in the clamp and adjacent pipe when subjected to clamp bolt load, tensile and compressive load on the clamp normal to the pipe. The Stresscoat evaluation was effective in determining the approximate stress distributions and the maximum stress locations for each type of loading. Based on these results detailed strain gage locations for subsequent testing were selected and presented.

- 1181 (ETEC-TDR--79-10) DYNAMIC TESTING OF A PROTOTYPICAL FFTF PIPING SYSTEM. Stearns, J.D.; Onesto, A.T.; Hunter, R.G. (Rockwell International Corp., Canoga Park, CA (USA). Energy Technology Engineering Center). 28 Sep 1979. Contract EY-76-C-03-0700. 257p. AT.

Dynamic tests were conducted on a fully insulated one-inch diameter prototypal FFTF pipe system to study the behavior of the pipe system and support structure during vibrational loading. Tests were conducted with both rigid struts for reference and mechanical snubbers. Test setup and sine sweep tests, simulated seismic tests, and a sine dwell to failure are described. Selected data are included in the Appendices.

- 1182 (ETEC-TDR--79-11) HEAD-FLOW MAPPING AND LOW NPSH TESTING OF A PROTOTYPIC SUBSCALE INDUCER PUMP AT SCTL. Haroldsen, D.D. (Energy Technology Engineering Center, Canoga Park, CA (USA)). 25 Sep 1979. Contract EY-76-C-03-0700. 180p. AT.

A series of tests have been conducted in sodium on a prototypic subscale inducer pump

designed by the Atomic International Division of Energy Systems Group. The tests, consisting of head-flow mapping and low NPSH operation were successful. The pump was removed from the SCTL test system and returned to Atomic International for inspection and preparation for future testing.

- 1183 (EUR--5598) EFFECTS OF FAST BREEDERS' CHARACTERISTICS ON CONSUMPTIONS AND EXPENDITURES RELATED TO ELECTRIC POWER GENERATION. AN ASSESSMENT FOR THE EUROPEAN COMMUNITY. Graziani, G.; Zanantoni, C. (Commission of the European Communities, Ispra (Italy). Joint Research Centre). Dec 1976. 36p. Dep. NTIS (US Sales Only), PC A03/MF A01.

The effects of the physical characteristics of the FBR (breeding ratio, inventory) and of the associated fuel cycle (out-of-pile time) on the results of strategy calculations performed for an electric power system are discussed. Attention is focused on the penetration of FBR, on the demand of uranium and separative work and on the economical performance of the breeders. It is pointed out that uncertainties about the out-of-pile time are as important as those concerning FBR breeding ratio and inventory.

- 1184 (EUR--5599) ROLE OF THE DEMAND-DURATION CURVE IN THE EVALUATION OF POWER STATION INSTALLATION POLICIES. REMARKS ON ITS EFFECTS ON STATIONS' UTILIZATION, GENERATING CAPACITY, LOAD FACTOR HISTORY AND CONSUMPTIONS. Graziani, G.; Maineri, M.; Zanantoni, C. (Commission of the European Communities, Ispra (Italy). Joint Research Centre). Dec 1976. 38p. Dep. NTIS (US Sales Only), PC A03/MF A01.

The role of the load-duration curve in determining the installation policy for an electric power system, is examined. The effect of the priority order for the allocation of the power stations in the load-diagram is also discussed: in the model used here (program TOTEM) this priority is given as a function of type and age of the station. The significance of the load factor history for a power station as a function of its age is also discussed. It is suggested that such a curve be deduced as a result of strategy calculations made with TOTEM rather than assuming it as input data.

- 1185 (EUR--5667(Pt.1), pp 455-475) FISSION-RATE MEASUREMENTS BY ACTIVATION TECHNIQUES: METHODS, ACCURACIES AND APPLICATIONS. Dudev, N.D.; Heinrich, R.R. (Argonne National Lab., Ill. (USA)). 1977.

From 1. ASTM-Euratom symposium on reactor dosimetry; developments and standardization; Petten, Netherlands (22 Sep 1975).

The U.S. reactor development programs have established the need for developing a capability to measure fission rates to an absolute accuracy of $\pm 2\%$. To achieve this goal, a national program is operative to develop and apply the necessary methodology. The approach is to use gamma-active fission product monitors to define fission rates for ^{235}U , ^{238}U , ^{237}Np and ^{239}Pu . Emphasis is upon redundant tests of the critical steps including mass assay, gamma counting, and fission yield determinations via a multiple laboratory approach. The methods are developed and tested in benchmark-type neutron fields, usually of low intensity ($<10^{12}$ n/cm²-sec) and applications of fission-rate measurements include EBR-II, fast critical assemblies, and CTR irradiation facilities. The most general approach, using conventional fission yields, is

capable of $\pm 3.5\%$ accuracy, whereas under certain specific conditions, accuracies of 2-2.5% are possible.

- 1186 (EUR--5667(Pt.1), pp 52-62) PREDICTION OF RADIATION DAMAGE IN STRUCTURAL MATERIALS OUTSIDE THE REACTOR CORE. Hehn, G.; Stiller, P. (Stuttgart Univ. (TH) (Germany, F.R.). Inst. fuer Kernenergetik und Energiesysteme). 1977.

From 1. ASTM-Euratom symposium on reactor dosimetry; developments and standardization; Petten, Netherlands (22 Sep 1975).

Computer studies were done to improve the prediction of radiation damage outside the reactor core for a PWR and a fast reactor. The aim was to show, how measurements of the neutron fluence and calculations of the energy spectra could be combined best. Outside the core there is a strong local variation of the neutron fluence accompanied by appreciable changes in neutron energy spectrum. With one- and two dimensional $S_{\text{sub}}(N)$ calculations the correlation was determined between surveillance samples at accelerated positions near the core and the expected damage in the pressure vessel. Displacement cross sections were calculated for the important nuclides in steel from ENDF/BIV data. The neutron flux spectra were weighted appropriately to get the displacement rates as a function of neutron energy. With this spectral quantity correlations are shown and possible uncertainties are discussed for both reactor types. Our results show, that calculations of displacement cross section and neutron flux spectra need high computational effort, if errors of 30% in the displacement rate sh.

- 1187 (EUR--5667(Pt.1), pp 375-413) CALCULATION OF ATOMIC DISPLACEMENT DOSE IN FAST REACTORS. Etherington, E.W. 1977.

From 1. ASTM-Euratom symposium on reactor dosimetry; developments and standardization; Petten, Netherlands (22 Sep 1975).

In the UKAEA neutron induced damage in metals is estimated in terms of atomic displacements. In the fast reactor environment this parameter is calculated from the fluxes obtained from multi-group neutron diffusion theory calculations and the Half-Nelson atomic displacement model. It is found that these calculated atomic displacements are sufficiently reliable in and near to the cores of fast reactors and it is not necessary to make regular experimental checks of the flux. An assessment has been made of the uncertainties in atomic displacements which arise from the atomic displacement model, the neutron cross sections of the material being damaged, the neutron spectrum, the relative total neutron flux, and the reactor power. These uncertainties have been compared with the requirement of the materials testing programme and of fast power reactor operation and it is judged that the present accuracy of the calculations meets these requirements. Some experiments which have a bearing on atomic displacement dose but which are being done primarily for other reasons are planned for the PFR. These measurements are reviewed.

- 1188 (EUR--5667(Pt.1), pp 99-116) NEUTRON SPECTROMETRY DATA IN LMFBR BENCHMARK AND STANDARD NEUTRON FIELDS. de Leeuw-Gierts, G.; de Leeuw, S. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)); Helmick, H.H.; Roger, J.W. 1977.

From 1. ASTM-Euratom symposium on reactor dosimetry; developments and standardization; Petten, Netherlands (22 Sep 1975).

The test of cross section data of dosimetry

materials, the study of the integral consistency of cross section files, as well as the determination of nuclear parameters, related to the development of fast reactors, and of material damage functions require a good spectral characterization of the neutron field. To reach the accuracy compatible with the requirements on the parameter to be deduced from integral measurements only theoretical treatment of the neutron field is inadequate and one experimental approach is by far insufficient. Sufficiently accurate neutron spectrum measurements as well as calculations can only be performed in clean assemblies. The assemblies suitable for these purposes are gathered in the family of benchmarks in which the neutron spectral characterization is in study. In Mol-II, Bigten and CFRMF the spectrum was among others determined by means of the ^6Li technique by the authors. Through these spectral measurements the link established between the German-Dutch-Belgian and the US Interlaboratory Reaction Rate (ILRR) programme in the field of fast reactor physics is enlarged.

- 1189 (EUR--5667(Pt.2), pp 46-60) DAMAGE ANALYSIS: DAMAGE FUNCTION DEVELOPMENT AND APPLICATION. Simons, R.L. (Westinghouse Hanford Co., Richland, Wash. (USA)); Odette, G.R. (California Univ., Santa Barbara (USA)). 1977.

From 1. ASTM-Euratom symposium on reactor dosimetry; developments and standardization; Petten, Netherlands (22 Sep 1975).

Material behavior observed under irradiation conditions in test reactors is not always directly applicable to the design of reactor components because of differences in the damage effectiveness of test reactor and service spectra. The interpolation and, under some conditions, extrapolation of material property change data to different neutron spectra from those in which the data were obtained can be performed using semi-empirical damage functions. The derivation and application of damage functions, including recent development, for the U.S. LMFBR and CTR programs, is reviewed. A primary application of damage functions is in predicting component life expectancies; i.e., the fluence required in a service spectrum to attain a specified design property change. An important part of the analysis is the estimation of the uncertainty in such fluence limit predictions. Additional uses of damage functions include improving the correlation of irradiation effects data, contributing to the understanding of irradiation effects mechanisms, and selecting future locations programs. The status of standardizing the procedures for the derivation and application of damage functions is discussed. Improvements in several areas of damage function development are needed before standardization can be completed. These include increasing the quantity and quality of the data used in the analysis, determining the limitations of the analysis due to the presence of multiple damage mechanisms, and finally, testing of damage function predictions against data obtained from material surveillance programs in operating thermal and fast reactors.

- 1190 (EUR--5667(Pt.2), pp 112-130) REACTOR DOSIMETRY INTEGRAL REACTION RATE DATA IN LMFBR BENCHMARK AND STANDARD NEUTRON FIELDS: STATUS, ACCURACY AND IMPLICATIONS. Fabry, A.; Ceulemans, H.; Vandeplas, P. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)); McElroy, W.N.; Lippincott, E.P. 1977.

From 1. ASTM-Euratom symposium on reactor

dosimetry; developments and standardization; Petten, Netherlands (22 Sep 1975).

CALIBRATION STANDARDS; DATA COMPILATION; IN PILE LOOPS; INTEGRAL CROSS SECTIONS; LMFBR TYPE REACTORS; NEUTRON DOSIMETRY; NEUTRON FLUENCE; NEUTRON FLUX; NEUTRON SPECTRA; NEUTRONIC DAMAGE FUNCTIONS; PHYSICAL RADIATION EFFECTS

- 1191 (EUR--5667(Suppl.), pp 23-89) REACTOR DOSIMETRY INTEGRAL REACTION RATE DATA IN LMFBR BENCHMARK AND STANDARD NEUTRON FIELDS: STATUS, ACCURACY AND IMPLICATIONS. Fabry, A.; Ceulemans, H.; Vandeplas, P. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)); McElroy, W.N.; Lippincott, E.P. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977.

From 1. ASTM-Euratom symposium on reactor dosimetry; developments and standardization; Petten, Netherlands (22 Sep 1975).

This paper provides conclusions that may be drawn regarding the consistency and accuracy of dosimetry cross-section files on the basis of integral reaction rate data measured in U.S. and European benchmark and standard neutron fields. In a discussion of the major experimental facilities CFRMF (Idaho Falls), BIGTEN (Los Alamos), $\Sigma\Sigma$ (Mol, Bucharest), NISUS (London), TAPIRO (Roma), FISSION SPECTRA (NBS, Mol, PTB), attention is paid to quantifying the sensitivity of computed integral data relative to the presently evaluated accuracy of the various neutron spectral distributions. The status of available integral data is reviewed and the assigned uncertainties are appraised, including experience gained by interlaboratory comparisons. For all reactions studied and for the various neutron fields, the measured integral data are compared to the ones computed from the ENDF/B-IV and the SAND-II dosimetry cross-section libraries as well as to some other differential data in relevant cases. This comparison, together with the proposed sensitivity and accuracy assessments, is used, whenever possible, to establish how well the best cross-sections evaluated on the basis of differential measurements (category I dosimetry reactions) are reliable in terms of integral reaction rates prediction and, for those reactions for which discrepancies are indicated, in which energy range it is presumed that additional differential measurements might help. For the other reactions (category II), the inconsistencies and trends are examined. The need for further integral measurements and interlaboratory comparisons is also considered.

- 1192 (EUR--5682) INTERPRETATION AND EXPERIMENT ON THE NEUTRON FLUX IN THE COOLING PIPES OF FAST REACTORS (A COMBINED METHOD SN-MONTE CARLO-ALBEDO FOR A RATIONAL CALCULATION OF THE RUPTURE). Bader, S. (Commission of the European Communities, Luxembourg). 1977. 104p. (In German). Dep. NTIS (US Sales Only), PC A06/MF A01.

The problems involved in the special case of SNR reactor are due to the structural dimensions, penetrations and tridimensional neutron transport through number of cavities. The previously available neutron codes separately dealt with the problems of penetrations and propagation in the event of rupture, but TIMOC cannot ensure precise computation and DLS/ALBEMO suffers a lack of reliability. A trial for a better approximation was performed; SN seems convenient for solving penetration problems and Monte Carlo for solving rupture. The thermal column was shown to play an important role (moderator plus reflector) and if it is not taken into account the epithermal flux obtained is not good. A group representation of iron cross sections is of importance for the flux spectrum after

filtering. In the rupture zone, the isotropy of the flux permits the albedo concept to give analytical representation but the relative dimensions geometry detectors does not allow analogy estimate in the spectra computation and, when the point of measurement is near the impact shock points: the method of estimate used caused some problems to arise. Results obtained show that the model such developed allow the experiments to be calculated in the limit of error, whilst the neglecting of the thermal column with TIMOC entailed bad results.

- 1193 (EUR--5813) REXCO-EUR CODE - USER'S MANUAL. Benuzzi, A.; Fasoli Stella, P. (Commission of the European Communities, Ispra (Italy). Joint Research Centre). 1978. 34p. Dep. NTIS (US Sales Only), PC A03/MF A01.

This report is the user's manual of a version (produced by the JRC and called REXCO-EUR) of the hydrodynamic-elastoplastic code REXCO-H release 2 developed by the ANL III.H USA. The REXCO-H code was designed for containment analysis of LMFBR during a high-energy excursion. The equations of mass, momentum and energy conservation in lagrangian coordinates for compressible fluid flow and the equations of state describing fluid materials are associated to the thin shell equations to calculate the dynamic response of a LMFBR to a core disruptive accident. The REXCO-EUR capabilities are described with the basic numerical techniques of the original code REXCO-H.

- 1194 (EUR--5854) SODIUM=FUEL INTERACTION: DROPPING EXPERIMENTS AND SUBASSEMBLY TEST. Holtbecker, H.; Schins, H.; Jorzik, E.; Klein, K. (Commission of the European Communities, Luxembourg). 1978. 58p. Dep. NTIS (US Sales Only), PC A04/MF A01.

Nine dropping tests, which bring together 2 to 4 kg of molten UO_2 with 150 l sodium, showed the incoherency and non-violence of these thermal interactions. The pressures can be described by sodium incipient boiling and bubble collapse; the UO_2 fragmentation by thermal stress and bubble collapse impact forces. The mildness of the interaction is principally due to the slowness and incoherency of UO_2 fragmentation. This means that parametric models which assume instantaneous mixing and fragmentation are of no use for the interpretation of dropping experiments. One parametric model, the Caldarola Fuel Coolant Interaction Variable Mass model, is being coupled to the two dimensional time dependent hydrodynamic REXCO-H code. In a first step the coupling is applicated to a monodimensional geometry. A subassembly test is proposed to validate the model. In this test rapid mixing between UO_2 and sodium has to be obtained. Dispersed molten UO_2 fuel is obtained by flashing injected sodium drops inside a UO_2 melt. This flashing is theoretically explained and modelled as a superheat limited explosion. The measured sodium drop dwell times of two experiments are compared to results obtained from the mentioned theory, which is the basis of the Press 2 Code.

- 1195 (EUR--5897, pp 221-231) STUDIES ON RECYCLING OF ACTINIDES OTHER THAN U AND PU. Bouchard, J.; Darrouzet, M.; Giacometti, A. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides). 1978. (In French).

From 1. technical meeting on the nuclear transmutation of actinides; Ispra, Italy (16 Apr 1977).

The destruction of actinides formed during the fuel cycle in electronuclear reactors is most effectively carried out by a recycling process in fast neutron reactors. This conclusion from a preliminary study has led to detailed calculations for americium, curium and neptunium. These computations were made on a twenty year operation time basis of a Super Phenix type fast reactor in which actinides associated with plutonium used as fuel were reintroduced. This experiment has shown that all these three elements can completely disappear without building up a great mass inventory in the reactor, without producing high quantities of heavier elements and with very limited reactor perturbations. Although the data required for these calculations are not well known yet and although experimental confirmations are needed, trends are sufficiently evident to give way to precise studies on possible cycles, including technological consequences, particularly on fuel fabrication and reprocessing.

- 1196 (EUR--5897, pp 13-50) REACTOR PHYSICS ASPECTS OF BURNING ACTINIDES IN A NUCLEAR REACTOR. Hage, W.; Schmidt, E. (Commission of the European Communities, Ispra (Italy). Joint Research Centre). 1978.

From 1. technical meeting on the nuclear transmutation of actinides; Ispra, Italy (16 Apr 1977).

A short review of the different recycling strategies of actinides other than fuel treated in the literature, is given along with nuclear data requirements for actinide build-up and transmutation studies. The effects of recycling actinides in a nuclear reactor on the flux distribution, the infinite neutron multiplication factor, the reactivity control system, the reactivity coefficients and the delayed neutron fraction are discussed considering a notional LWR or LMFBR as an Actinide Transmutation Reactor. Some operational problems of Actinide Transmutation reactors are mentioned, which are caused by the α -decay heat and the neutron sources of Actinide Target Elements.

- 1197 (EUR--5897, pp 193-212) SOME PRELIMINARY INVESTIGATIONS OF THE INCINERATION OF ACTINIDE WASTE FROM THE UK AND EUROPEAN NUCLEAR POWER PROGRAMMES. Harte, G.A.; Nair, S. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). 1978.

From 1. technical meeting on the nuclear transmutation of actinides; Ispra, Italy (16 Apr 1977).

The reactor inventory codes FISP /RICE and HYACINTH have been used to predict waste arisings from the operation of various reactor types, and an investigation has been carried out to determine the effect on the long-term hazards of these wastes of the removal and nuclear incineration of the elements neptunium, americium and curium.

- 1198 (EUR--5946) COMPARATIVE ANALYSIS OF A HYPOTHETICAL COOLANT LOSS ACCIDENT IN AN LMFBR REACTOR WITH THE USE OF VARIOUS CALCULATION MODELS FOR A COMMON REFERENCE PROBLEM. Royl, P. (Commission of the European Communities, Brussels (Belgium)). 1979. 202p. (In French). Dep. NTIS (US Sales Only), PC A06/MF A01.

The results of a comparative analysis of the initial and dismantling stages of a hypothetical loss of flow accident in an LMFBR reactor are presented. The analyses were made for a common reference problem with four different calculation models (CARMEN/KADIS,

SURDYN, CAPRI/KADIS and FRAX). The reference core is described specifically, as are the differences in its geometrical disposition in the models, the static and transient conditions before and after the start of boiling and during dismantling. The differences in the models used for simulating the boiling and the dismantling are compared. The structure of the core, as well as the calculation conditions and hypotheses, were intentionally designed so that the accident would culminate, in all cases, in an energetic hydrodynamic dismantling stage.

- 1199 (EUR--6108, pp 53-58) MEASUREMENTS AND CALCULATIONS OF INTEGRAL CAPTURE CROSS-SECTIONS OF STRUCTURAL MATERIALS IN FAST REACTOR SPECTRA. Seth, S.; Brunson, G.; Gmuier, K.; Jermann, M.; McCombie, C.; Richmond, R.; Schmocker, U. (Eidgenössisches Inst. fuer Reaktorforschung, Wuerenlingen (Switzerland)). 1979.

From Meeting on neutron data of structural materials; Geel, Belgium (5 Dec 1977).

This paper relates the checking of integral data of steel and iron in fast reactor lattices. The fully-rodged GCFR benchmark lattice of the zero-energy reactor PROTEUS was successively modified by replacing the $\text{PuO}_2\text{-UO}_2$ fuel rods by steel-18/8 or steel-37 (iron) rods. The neutron spectra of the modified lattices in fact have median energies close to that of a typical LMFBR. The replacement of fuel by the structural material of interest was such that in each case the value of $k(\infty)$ was reduced to near-unity. This allowed the measurement of the lattice- $k(\infty)$ by the null-reactivity technique. In addition, the principal reaction rates (namely U238 capture and fission, relative to Pu239 fission) and the neutron spectrum were measured. These directly measured integral data which are particularly sensitive to the steel cross-sections can be used for the checking and systematic adjustment of data sets. The results may also be analysed so as to derive specific values for the integral capture cross-sections of steel and iron. Neutron balance equations were set-up for each of the lattices using the measured $k(\infty)$ and reaction rates.

- 1200 (EUR--6108, pp 105-117) BENCHMARK TESTS OF JENDL-1 FOR STRUCTURAL MATERIALS. Otake, I. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). 1979.

From Meeting on neutron data of structural materials; Geel, Belgium (5 Dec 1977).

In order to examine the reliability of JENDL-1 for its use of FBR core design, various benchmark tests were performed on the basis of selected integral measurements. Total of 21 assemblies were selected and sample worths were calculated for ^{10}B , Cr, Fe, Ni, C, Na, Al and Mo. The adjustment technique was applied for the cross sections to fit the experimental values. Besides, fast neutron spectra in Fe were calculated and then compared with the measurements at KURRI to evaluate the JENDL-1 data for structural materials.

- 1201 (EURFNR--1454) SODIUM BOILING EXPERIMENTS IN AN ANNULAR TEST SECTION UNDER FLOW RUNDOWN CONDITIONS. Kaiser, A.; Pepler, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). Mar 1977. 44p. (KFK--2389). Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States--Euratom Fast Reactor Exchange Program.

Experiments have been carried out using sodium in an annular test section geometry to

simulate boiling development in a fast reactor core under the somewhat improbable conditions of primary circulator pump run-down with coincidental failure of the reactor trip system. Induction heating of the outer tube was used, the axial heat flux distribution being linear. Boiling was achieved by means of a steady reduction of the coolant flow from normal steady state values, heat fluxes being appropriate to reactor conditions. Boiling was allowed to continue until indications of permanent dryout of the heated tube were received, at which stage power was terminated to prevent damage to the test section. The results obtained provide useful information on the transient development of boiling, the axial distribution of void within the channel, initial superheat, liquid film thickness, dryout and rewetting. The experiments thus provide a source of data against which computer codes such as BLOW3 based on theoretical models of boiling may be validated.

- 1202 (EURFNR--1457) PHYSICS INVESTIGATIONS OF SODIUM-COOLED FAST REACTORS: SNEAK-ASSEMBLY 9C. Scholtyssek, W. (comp.). (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Apr 1977. 81p. (KFK--2361). Dep. NTIS (US Sales Only), PC A05/MF A01.

Work performed under United States--Euratom Fast Reactor Exchange Program.

The assemblies 9C completed the SNEAK 9 series of SNR specific criticals. A detailed description is presented of the SNEAK 9C-assemblies and of experiments related to core physics and to neutron physics parameters of the compositions used. The results of the experiments are compared to calculations carried out mainly with Karlsruhe code systems and the KFKINR cross section set. Only in some cases other cross sections were used.

- 1203 (EURFNR--1460) THERMODYNAMIC STATE, SPECIFIC HEAT, AND ENTHALPY FUNCTION OF SATURATED UO_2 VAPOR BETWEEN 3000 K AND 5000 K. Karow, H.U. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Feb 1977. 40p. (KFK--2390). Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States--Euratom Fast Reactor Exchange Program.

Reactor safety analysis requires knowledge of the thermophysical properties of molten oxide fuel and of the thermal equation-of-state of oxide fuel in thermodynamic liquid-vapor equilibrium far above 3000°K. The thermodynamic state of saturated UO_2 fuel vapor, its internal energy $U(T)$, specific heats $C_{\text{sub}} v(T)$ and $C_{\text{sub}} p(T)$, and enthalpy functions $H^0(H)$ and $H^0(T)-H^0_0$ have been determined by means of statistical mechanics in the temperature range 3000 to 5000°K. The discussion of the thermodynamic state includes the evaluation of the plasma state and its contribution to the caloric variables-of-state of saturated oxide fuel vapor. The resulting enthalpy functions and specific heats for saturated UO_2 vapor of equilibrium composition and that for pure UO_2 gas are compared with the enthalpy and specific heat data of gaseous UO_2 at lower temperatures known from literature. Tentatively also experimental enthalpy data of the condensed phases of UO_2 have been converted into the vapor phase using the Clausius-Clapeyron equation. However, the correlation based on the Clausius-Clapeyron equation turns out to fail because this equation is not applicable to the chemically reactive oxide fuel system.

1204 (EURFNR--1464) ANALYSIS OF EXPERIMENTAL FISSION GAS BEHAVIOR DATA IN FAST REACTOR FUEL UNDER STEADY STATE AND TRANSIENT CONDITIONS. Fischer, E.A. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Jun 1977. 35p. (KFK--2370). Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States--Euratom Fast Reactor Exchange Program.

To assess the utility of the computer programs LANGZEIT and KURZZEIT for the prediction of the fission gas behavior in LMFBR accident analysis, the input parameters were reassessed, a systematic comparison of the results with experimental data was carried out, and the model was further developed to include the diffusion of gas bubbles in a thermal gradient. The empirical relation by Duit was used for a comparison of the predicted gas release with experimental data. In addition, a comparison with data obtained in the fuel irradiation program of the Debenelux Fast Breeder Project was also carried out. The LANGZEIT results obtained using the new input parameters are consistent with experimental data for burn up values above 20,000 MWd/to. For lower burn up, LANGZEIT overestimates the gas release, because it considers the gas at grain boundaries as released. (An improved model is now available.) The program KURZZEIT was used to analyze transient gas release laboratory experiments carried out at HEDL. For this purpose, the effect of bubble diffusion in a thermal gradient had to be included in the model, because it plays an important role in those experiments. If reasonable values for the bubble diffusion coefficient are used, the model predicts correctly the time needed until complete gas release in the HEDL experiment, but not the details of the release history.

1205 (EURFNR--1465) IMPORTANCE AND STATUS OF (n,α) -CROSS SECTIONS FOR A RELIABLE PREDICTION OF RADIATION DAMAGE IN STAINLESS STEEL. Goel, B. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Jun 1977. 29p. (KFK--2473). Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States--Euratom Fast Reactor Exchange Program.

It is well established that helium formed in stainless steel by various (n,α) processes has a pronounced effect on its mechanical and dimensional properties. There has been strong discrepancy between the helium contents measured in irradiated stainless steel and that calculated on the basis of known (n,α) cross sections on the stable constituents of stainless steel. This discrepancy seems to be resolved by attributing the excessive helium to the two-step process $^{58}\text{Ni}(n,\gamma)^{59}\text{Ni}(n,\alpha)^{56}\text{Fe}$. However, a large discrepancy exists between the value of the thermal cross section for the $^{59}\text{Ni}(n,\alpha)^{56}\text{Fe}$ process used to calculate the helium production data and that obtained from the direct measurements of this cross section. The role of the (n,α) cross section on the irradiation-induced swelling is investigated. It is shown that insoluble gases like helium do influence swelling by way of nucleating and stabilizing voids. It is further shown that helium produced via ^{59}Ni constitutes a substantial amount of helium produced in stainless steel under thermal as well as under fast reactor conditions. In fast reactors helium produced via ^{59}Ni gains its importance at high neutron fluences. A linear extrapolation of helium production data measured at low fluences to the fluences of the

order of 1.5×10^{23} n/cm² is not justified. It is shown that the usual practice to relate helium production data to thermal and fast neutron fluence is inadequate. The details of neutron spectrum are necessary to predict helium production rate reliably. A measurement of $^{59}(n,\alpha)^{56}\text{Fe}$ cross section in keV range is recommended. 8 figures, 4 tables.

1206 (FBR--77-4) PLBR PHASE II FOLLOW-ON RESOLUTION OF KEY ISSUES. (Atomics International Div., Canoga Park, CA (USA)). Sep 1977. Contract EV-76-C-03-1142. 407p. AT.

Results of the effort to resolve principal outstanding engineering problems and issues arising in incompleting areas of systems conceptual design during Phase II of the Prototype Large Breeder Reactor (PLBR) Design Project are presented. Tasks covered under this effort included steam cycle comparison, in-vessel primary pipe inspection and maintenance, reactor cover, vessel and internals seismic response, primary system piping arrangement and analysis, in-vessel flow distribution, fuel handling system, intermediate piping system, reactor upper internals, EVST and cooling system, and plant control.

1207 (FBR--77-4) PLBR PHASE II FOLLOW-ON STUDIES: RESOLUTION OF KEY ISSUES. (Atomics International Div., Canoga Park, CA (USA)). Sep 1977. Contract EV-76-C-03-1142. 369p. AT.

Results of the effort by Atomics International to resolve principal outstanding engineering problems and issues arising in incompleting areas of systems conceptual design of Phase 2 of the Prototype Large Breeder Reactor Design Project are presented. Tasks covered include: steam cycle comparison; in-vessel primary pipe inspection and maintenance; reactor cover; vessel and internals seismic response; primary system piping arrangement and analysis; in-vessel flow distribution; fuel handling system; intermediate piping system; reactor upper internals; EVST and cooling system; and plant control.

1208 (FEI--536) HIGH-TEMPERATURE ACOUSTIC TRANSDUCERS OF SOUND PRESSURE BASED ON LITHIUM NIOBATE PIEZOELEMENTS. Masagutov, R.F.; Krivtsov, V.A. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). 1975. 22p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

Reviewed are wide-band piezoelectric sensors of sound pressure for measuring characteristics of sound fields in a liquid-metal coolant at temperatures up to 800 deg C. Materials used provide the possibility of using the sensors in the ionizing radiation fields. Two types of the sensors, i.e. asymmetric, are described. Their electrical, mechanical and acoustic parameters are presented. Thermal stability of the sensors is tackled. Estimations show that a higher temperature does not affect largely sensor characteristics, except for electric resistance which drops down to $10^6\Omega\text{cm}$ as the temperature goes up to 600 deg C.

1209 (FEI--556) EXPERIMENTAL STUDY ON INTERCHANNEL INTERACTION OF COOLANT IN ROD LATTICES WITH WIRE-WRAP SPACERS. Zhukov, A.V.; Kotovskij, N.A.; Matyukhin, N.M.; Sviridenko, E.Ya.; Sorokin, A.P.; Ushakov, P.A. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). 1975. 26p. (In

Russian). Dep. NTIS (US Sales Only), PC A03/MF A01.

Experimental study results for interchannel mixing in rod assemblies with wire spacers are presented. Particular attention is given to the analysis of interchannel interaction local processes found by measuring temperature profiles along the assembly length. The results of experimental study are compared with precomputed data on the basis of local and effective interchannel mixing models. A systematization is provided of data on the effective mixing coefficients for the rod assemblies in the liquid metal flows. Relations for local and effective mixing coefficients are recommended.

- 1210 (FEI--563) DESCRIPTION OF DESIGNS, FABRICATION AND TEST DATA OF PIEZOELECTRIC PICK-UPS BASED ON LITHIUM NIOBATES. Masagutov, R.F.; Krivtsov, V.A. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). 1975. 14p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

Piezoelectric sensors of sound pressure are discussed. These are designs on the basis of piezoelements of lithium niobate and intended for application in power units, in particular, in the fast nuclear reactor core. Two types of the sensors, i.e. asymmetric and symmetric, are described. Problems of sensor manufacture, assembly and their preliminary treatment are considered, which ensures their reliability. Results of sensor tests in the air furnace and in the unit with a forced circulation of Na-K alloy show that the sensors provide a reliable output signal at temperatures up to 700 deg C.

- 1211 (FEI--566) EXPERIMENTAL-CALCULATION STUDIES ON EFFICIENCY OF EUROPIUM OXIDE RODS OF THE BFS-29 ASSEMBLY. Gorbato, N.E.; Matveev, V.I.; Tyutyunnikov, P.L.; Efimenko, E.M.; Efimov, E.P.; Ivanov, A.P.; Melamed, V.E.; Abagyan, L.P.; Zakharova, S.M. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). 1977. 17p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

Experimental results of measurements of the efficiency of the rods made of natural boron carbide and europium oxide taken on one of the models of the BN-600-reactor - BFS-29 assembly are presented. The experimental data are compared with the results of calculations. It is shown that the experimental data agree (within 10%) with the calculated results. The analysis of the obtained results shows that the efficiency of the identical rods made of natural boron carbide and europium oxide is approximately equal in the fast power reactors, type BN-600.

- 1212 (FEI--597) APPROXIMATE SIMULATION TECHNIQUE FOR THERMAL AND HYDRAULIC PARAMETERS OF A REACTOR CORE. Gorchakov, M.K.; Kolmakov, A.P.; Yur'ev, Yu.S. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). 1975. 19p. (In Russian). Dep. NTIS (US Sales Only), PC A03/MF A01.

Differential equations are given of liquid motion and heat transfer in the core as in a porous body in case of natural and forced convection. Conditions of macrodistribution similarity of liquid pressure, velocity and temperature are determined. An example is shown simulating quasistationary processes in the liquid-metal reactor with a hydromodel.

- 1213 (FEI--607) EFFECT OF THE FAST BREEDER CONCEPT ON THE REACTOR MAIN CHARACTERISTICS AND STEAM GENERATION PARAMETERS. Rinejskii, A.A.; Pashicheva, M.I.; Chushkin, V.N. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). 1975. 29p. (In Russian). Dep. NTIS (US Sales Only), PC A03/MF A01.

To solve a fuel problem for a particular country fast breeders should have definite physical characteristics (reproduction coefficient, duplication time). If this condition is taken as a basic one for manufacture of a fast reactor, it causes a number of specific requirements for the reactor, thermomechanical and power-steam equipment. The above problems are discussed.

- 1214 (FEI--622) ANALYTICAL SOLUTION OF SOME PROBLEMS RELATED WITH ELECTROMAGNETIC METHOD OF VELOCITY MEASUREMENT IN COMPLEX CHANNELS. Mikhin, V.I.; Zhukov, A.V. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). 1975. 20p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

The objective of the work is to study a process of measuring the velocity of liquid-metal coolant by an electromagnetic method. The operation of electromagnetic sensors, which principle lies in a EMF effect caused in a rotating magnetic field of a conductor, has been analytically studied. The examination of two types of the electromagnetic sensor, one with and the other without a copper ring, is aimed at determining the effect of channel geometry and location of electrodes on the sensor signal. The analysis is performed for the case of a flat flow of the coolant. The formulas are recommended describing the effect of adjacent walls of various configurations on the signal of the sensor. The following conclusions have been drawn: in case of a flat flow of the coolant the effect of the adjacent walls of the channel on the sensor signal is largely governed by a size of the magnet, the use of a copper ring in the electromagnetic sensor makes it possible to diminish the effect of the adjacent walls of the channel upon the sensor signal, application of small-size magnets improves considerably the characteristics of electromagnetic sensors.

- 1215 (FEI--624) ABOUT SOME EFFECTS ON THE COOLANT TEMPERATURE FIELD IN FUEL ASSEMBLIES WITH WIRE SPACERS. Zhukov, A.V.; Sorokin, A.P.; Titov, P.A.; Ushakov, P.A. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). 1975. 21p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

The effect of several factors - displacement of fuel elements, excess energy output from central and peripheral fuel elements, spacing of spacer coils, etc. - on the temperature field of fast reactor fuel assemblies was examined. Methodology for calculating the heat field of reactor BN-600 with consideration of interchannel interdisplacement is presented. Analysis of the calculation results showed that variations in the fuel element lattice of fast reactors from the normal geometry, within permissible limits, on the assembly and also excess (up to 50%) energy release from individual fuel elements shows up somewhat in average coolant heating along the channels. The effects increase significantly with displacement of the peripheral fuel elements. Placement of detectors in the cell channel

significantly affect the distribution of the coolant heating. Dispersion of the coolant heating along the cell radius is shown graphically as a function of the factors studied.

- 1216 (FEI--711) TEMPERATURE CONTROL OF ACTIVE CORE ASSEMBLIES IN A FAST REACTOR IN ACCIDENTS. Buksha, Yu.K.; Bagdasarov, Yu.E.; Kuznetsov, I.A. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). 1976. 24p. (In Russian). Dep. NTIS (US Sales Only), PC A03/MF A01.

A mathematical model of nonstationary two-dimensional temperature field in a fast reactor active core, when the coolant flow rate change in the separate fuel assemblies takes place. On the base of calculations the correction factors for indications of thermocouple located at fuel assembly output have been obtained. Effectiveness of using in reactor protective system thermocouples with various constants of time and flowmeters controlling each fuel assembly state has been investigated. Specific calculation results are related to BN-600-type reactor.

- 1217 (FEI--739) WAYS OF SOLVING SOME ENGINEERING PROBLEMS IN ARRANGEMENT OF INTEGRAL AND LOOP TYPE FAST REACTORS. Rineiskii, A.A.; Baklushin, R.P.; Ivanenko, V.N.; Klimov, S.A.; Mel'nikov, N.P.; Matveev, G.V.; Sobolevskii, R.G.; Tkachenko, S.B.; Ustinov, G.G. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). 1977. 21p. (In Russian). Dep. NTIS (US Sales Only), PC A03/MF A01.

A comparative analysis of currently-in-use and feasible arrangements of loop and integral LMFBR type reactors is given. The discussion is based on the designs of "in operation", "under construction" and "projected" reactors all over the world. The effects of the outlet coolant temperature, reactor unit power, maintainability and costs on the choice of the arrangement type are considered. Considered in greater details are the hydraulic schemes and problems of the integral reactor vessel attachment as well as problems of ensuring strength of pipe-to-vessel connections and localizing coolant leaks in the loop type reactors. There is no general consent over the superiority of one of the designs up to now. There are different opinions on the account in different countries, since both designs have its merits and shortcomings.

- 1218 (FEI--850) PHYSICAL ASPECTS OF CHOOSING FUEL FOR A RESEARCH FAST REACTOR. Antonova, V.N.; Lantsov, M.N.; Markelov, I.P.; Orehov, V.I.; Chufikov, M.I. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). 1978. 10p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

Comparative evaluations of research sodium cooled fast reactors fueled with plutonium or uranium carbides have been made. The core represents a homogeneous mixture of major constituents - fuel, sodium and steel. The physical calculations have been fulfilled in terms of the 26-group P_1 -approximation in the spherical geometry within the wide range of core volumes (20-1000 l). The plutonium fuel is concluded to be superior to the uranium one from the standpoint of the reactor "quality" (ratio of the maximum neutron flux to the reactor power), fuel concentration in the critical volume of the core and the neutron

spectrum hardness.

- 1219 (FRNC-TH--709) STUDY OF SODIUM COMBUSTION AND FIRE EXTINCTION BY PULVERIZED SUBSTANCES. ROLE OF ADDITIVES. Reuillon, M. (Orleans Univ., 45 (France)). 1976. 153p. (In French). Dep. NTIS (US Sales Only), PC A10/MF A01.

Thesis.

A study is presented on inflammation and combustion of liquid sodium, extinction of the metal fires by comburant concentration reducing and cooling, liquid covering, powder smothering. The role of the additives is discussed. The setting up and the experimental process are given. The sodium combustion residues are analyzed. Various powder mixtures based on alkaline carbonates, $\text{NaCl}-\text{Na}_2\text{CO}_3$, $\text{NaCl}-\text{Na}_2\text{CO}_3, \text{H}_2\text{O}$ etc... are studied. An attempt of interpretation on sodium fire extinction is presented.

- 1220 (FRNC-TH--765) STUDY OF SODIUM BOILING DURING A POWER TRANSIENT IN A HEATING DUCT. INVESTIGATION CARRIED OUT WITHIN THE SCOPE OF A STUDY ON THE SAFETY OF FAST NEUTRON REACTORS. Seiler, J.M. (Grenoble-1 Univ., 38 (France); Institut National Polytechnique, 38 - Grenoble (France)). 1977. vp. (In French). Dep. NTIS (US Sales Only), PC A10/MF A01.

Thesis.

This work comprises an experimental study and a physical and theoretical interpretation of sodium boiling during a power transient in conditions simulating a power excursion accident in a breeder reactor. The experimental study was performed on an 'out of pile' system with forced sodium convection. It enabled a detailed study to be made of the draining of the heating duct representing a part of a reactor assembly (expansion of the double-phase area on the heating part of the pin). The physical and theoretical interpretation rests on a very simple calculation model solely describing the heat transfers in the duct, assuming that the fluid stays liquid. The extension of the boiling is governed by the initial overheating and the boiling conditioned by thermal inertias of the walls. The extent of the overheating appears to depend on the degree of initial under-saturation. In the absence of overheating, the rate at which the double-phase front moves is controlled by the saturated boiling. This front is well localised when the central temperature gradient is significant. The draining rate is then directly proportional to the growth rate of the temperature in the liquid. The model employed in the CASPAR code for describing power excursion experiments in single liquid phase is described. This code makes it possible to calculate the heat transfers occurring in a forced system by a heating part, a coolant in movement and possibly a part representing an outer cover.

- 1221 (FRNC-TH--783) STUDY OF TRANSIENT AND PERMANENT FLOW IN THE EVENT OF NATURAL CONVECTION IN A CONFINED ENVIRONMENT. Tenchine, D. (Lyon-1 Univ., 69 (France)). 1978. vp. (In French). Dep. NTIS (US Sales Only), PC A11/MF A01.

These (D. Ing.).

This report deals with natural convection in a confined environment, in connection with the studies on the safety of nuclear reactors of the sodium cooled breeder type (possibilities of removing the residual power of the fuel by natural convection in the liquid sodium). These natural convection exchanges develop in a confined environment between various sodium volumes separated by metallic structures. The

study covered a cavity heated by the roof or by the bottom and cooled laterally. The results are compared with those achieved along heating plates, vertical or horizontal, in an infinite medium and the effect of the thermal limit conditions are highlighted by comparison with the case of bottom heated and roof cooled cavities. Placed in a bidimensional geometry situation, with water as fluid, this leads to tackling the problems of similarity between water and sodium flows. A digital code has been developed in plane bidimensional geometry with a laminar and permanent flow. A description is given of the 'BIDIM' experimental rig as well as the measuring and display devices. A permanent flow study of the two previously mentioned configurations produces references for the analysis of transient flows, particularly in the case of bottom heating (field of medium temperatures and medium exchange coefficient). The turbulence intensity and frequency distribution determinations of the temperature changes are given. Then the determinations in transient flow are dealt with in the case of bottom heating. The cavity being initially cold, a power rise is initiated in the heating plates and the establishment and growth of natural convection and the change in the field of medium temperatures and exchange coefficient are studied.

- 1222 (FWC/FWEC/ND--75-70) 360° FLOW MODEL PRIMARY INLET FLOW DISTRIBUTION TEST REPORT. Dietz, D. (Foster Wheeler Energy Corp., Livingston, NJ (USA)). 15 Dec 1975. Contract EY-76-C-15-0003. 146p. Dep. NTIS, PC A07/MF A01.

In this test, an appropriate hole pattern in the distribution shroud was obtained so as to achieve uniform shellside sodium flow at a full load condition. The flow distribution through the distribution shroud was measured at lower flow rates after the final hole pattern was determined. The velocity profile inside the nozzle and in the inlet plenum was also measured over a range of flow rates. The pressure drops upstream of the distribution shroud and across the distribution shroud were also measured. A visual study to obtain a clear picture of the flow patterns in the inlet plenum completed the test program.

- 1223 (FWC/FWEC/ND--76/49) INTERMEDIATE FLOW MODEL TEST REPORT. Dietz, D. (Foster Wheeler Energy Corp., Livingston, NJ (USA)). 27 Sep 1976. Contract EY-76-C-15-0003. 91p. Dep. NTIS, PC A05/MF A01.

The intermediate flow enters the top of the Clinch River Breeder Reactor Intermediate Heat Exchanger (CRBRP-IXH), passes downward through the downcomer pipe and turns 180° in the plenum formed by the intermediate hemi-head and the tubesheet. The flow then passes through the tube bundle and exits the heat exchanger. Due to the geometry of the plenum, the flow tends to be strongest in the tubes along the outer perimeter of the tube bundle. The tests determined the geometry of an annular ring baffle secured to the wall of the hemispherical head which deflects the tubeside flow evenly into the entrance region of the tube bundle, and the length of the downcomer extension.

- 1224 (FWC/FWEC/ND--76/59) CRBRP-IXH: 30° MODEL FLOW DISTRIBUTION, PRESSURE DROP AND TUBE VIBRATION TEST REPORT. Jaisingh, G. (Foster Wheeler Energy Corp., Livingston, NJ (USA)). 1 Mar 1977. Contract EY-76-C-15-0003. 133p. Dep. NTIS, PC A07/MF A01.

The objective of this test program was to study the flow of shell side sodium and tube

vibration caused by it. This was achieved by simulating sodium flow by water at 170°F. (Additional tests were performed with 100°F water at lower flow rates.)

- 1225 (GEAP--13923-16) SAFETY ENGINEERING SIXTEENTH QUARTERLY REPORT, MAY--JULY 1976. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). Aug 1976. Contract E(04-3)-0893-TAS-2. 57p. AT.

Progress is summarized in the following LMFBR safety studies: experiment analysis and code validation, evaluation and adaptation of existing codes, fuel performance code development, and parametric accident analysis. (DG)

- 1226 (GEFR--00027(L)) INTERIM EVALUATION OF SODIUM--WATER REACTION TEST NO. 1 DATA AND COMPARISON WITH TRANSWRAP ANALYSES: SERIES I LARGE LEAK TEST PROGRAM. Amos, J.C.; McClelland, S.D.; Regimbal, J.J. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). 28 Feb 1977. Contract EY-76-C-03-0893-010. 65p. AT.

The report presents an interim evaluation and analytical comparison of the test data and intertest examination results for Sodium Water Reaction (SWR) Test No. 1 (SWR No. 1) conducted in the Large Leak Test Rig (LLTR) at the Liquid Metal Engineering Center (LMEC). Efforts to achieve a more detailed evaluation of SWR No. 1 are underway. Comparisons with the experimental findings of subsequent tests in the series, similar to this evaluation, are planned. A final Series I evaluation report will be prepared following completion of the Series I test program. The test was the first of the Series I large leak tests utilizing the MSG test article. The test consisted of a programmed double-ended guillotine rupture of a centrally located steam generator tube at a location 4'10³/' above the MSG lower tubesheet. Pretest conditions for both water and sodium in the test article were representative of CRBRP evaporator operating conditions. Dynamic H₂O flow conditions were established in the rupture tube, with static (uniform) steam pressure in the remaining steam generator tubes. A representative 225° linear temperature gradient was established over the length of the shell.

- 1227 (GEFR--00037(L)) LLTR SERIES I: ANALYSIS IN SUPPORT OF SWRPS DESIGN AND SWR CLEANUP AND RECOVERY. Reynolds, R.G. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). Apr 1977. Contract EY-76-C-03-0893-010. 29p. AT.

Documentation and analysis is made of the Series I Large Leak Test data pertinent to the CRBR Sodium-Water Reaction Pressure Relief System (SWRPS) and the Sodium-Water Reaction (SWR) Cleanup and Recovery Programs. This is an interim report and it will be updated when all the results from SWR-2 and SWR-3 are available. Preliminary data reveal no CRBRP rupture disc design flaws. The long vent lines in the LLTR are effective in reducing the amount of sodium compound aerosols released to the atmosphere so there is no incentive to test a centrifugal separator in Subsequent Series II tests. Enough reaction products were deposited in the steam generator during the test to present a significant clean-up problem. Reaction products were not observed to stick to relief system lines.

- 1228 (GEFR--00041) PROBABILISTIC LEAK ANALYSIS FOR LMFBR PRIMARY COOLANT PIPING.

Derby, S.L.; Hackford, N.E. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jul 1977. Contract EY-76-C-03-0893 013. 32p. AT.

The development of a probabilistic analysis for a leak in the primary coolant piping in an LMFBR is described. In a detailed example, the analysis is used to evaluate a point estimate of the conditional probability of pipe wall penetration as a function of initial crack depth for the inlet elbow in the reactor primary coolant system of the Fast Flux Test Facility (FFTF). The result is used to demonstrate an approach for specifying a probabilistic criterion for the crack inspection of piping to meet an overall safety objective.

1229 (GEFR--00051(L)=1) SELF-WASTAGE PHENOMENA. VOLUME I. STATE-OF-THE-ART DATA REVIEW AND EVALUATION. Greene, D.A. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1977. Contract EY-76-C-03-0893-010. 86p. AT.

The report discusses general trends and conclusions which can be drawn from self-wastage data, and supports the evaluation with hypotheses and mechanisms on self-wastage and defect plugging phenomena. The data base is used to provide preliminary estimates of the magnitude of self-wastage phenomena, their impact on leak detection criteria, and possible criteria and guidelines for the plant operator. Chemical leak detectors were evaluated against the self-wastage data. They provide significant, but limited, protection against defects in the steam generators. Such detectors will indicate the presence of defects which have an initial size in the range 0.025 to 0.10 mm equivalent hydraulic diameter (0.001 in. to 0.004 in.). If the defect is initially smaller than 0.025 mm diameter and in the superheater (460°C), it is probable that the defect will grow into a hole of approximately 3 mm diameter prior to unit shutdown following detection and operator action. All defects with initial size below 0.2 mm (0.008 in.) in the evaporator (340°C) will probably be detected with time to take action. Acoustic leak detectors were also evaluated. They appear to have the potential for increased detection range compared to chemical detectors, but are in an early stage of development. Acoustic detectors have the potential added capability of location of the defect.

1230 (GEFR--00051(L)=2) SELF-WASTAGE PHENOMENA. VOLUME II. EXPERIMENTAL DATA FROM GE TEST FACILITIES. Gudahl, J.A.; Greene, D.A.; Hunsicker, J.C.; Ulpindo, C.Q. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Feb 1977. Contract EY-76-C-03-0893-010. 179p. AT.

A study on self-wastage phenomena in LMFBR steam generators is described. Experimental data on preliminary scoping tests and the Phase I test series are included. The report also contains descriptions of the test facilities in which the tests were made and a description of the techniques used to obtain the data.

1231 (GEFR--00061(L)) SOME ASPECTS OF CORROSION OF 2 1/4 CR--1 MO EVAPORATORS BY PROTOTYPIC CRBRP WATER CHEMISTRY. Rodgers, D.N.; Dutina, D. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Mar 1977. Contract EY-76-C-03-0893-010. 10p. AT.

The steam/water tubes in the Clinch River Breeder Reactor (CRBR) evaporators will experience Departure from Nucleate Boiling

(DNB) in which the liquid film in the nucleate boiling region evaporates and breaks up into rivulets and dry spots. A number of potential damage mechanisms have been postulated as a result of this. For example, a major concern has been that tube life would be reduced by fatigue failure due to thermally induced stresses. Such stresses arise from oscillation in the tube wall temperature caused by the repeated wetting and drying within the DNB zone and the change from nucleate to film boiling as the DNB zone moves along the tube. Recent sophisticated modeling work coupled with experimental data has virtually eliminated this concern. It is now accepted that the CRBR design provides adequate margin against fatigue failure for the design life at the worst operating condition (64 percent power). A second possible damage mechanism was that corrosive species, particularly sodium hydroxide, would concentrate in the dry-out region and cause local corrosion damage. This paper emphasizes the work carried out to minimize this potential problem and to test its severity.

1232 (GEFR--00062) SWELLING OF TYPES-304, -316, AND -321 STAINLESS STEELS AT HIGH FLUENCES IN EBR-II. Kenfield, T.A.; Appleby, W.K.; Bell, W.L.; Busboom, H.J.; McClellan, G.C. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Mar 1977. Contract EY-76-C-03-0893-003. 56p. AT.

Swelling determinations on unstressed cladding tubes have revealed peak swellings of 14% and 18%, respectively, for solution-annealed Types-304 and -316 after irradiation to 14×10^{22} n/cm² ($E > 0.1$ MeV). The swelling peaks occurred at approx. 500°C and approx. 580°C. The 20% cold-worked Type-316 showed a slight reduction in swelling (15%) relative to solution-annealed material, although the temperature dependence of swelling remained essentially the same (peaking at approx. 580°C). The peak steady-state swelling rate for 20% cold-worked Type-316 stainless steel, the reference cladding and duct material for FFTF and CRBRP, is approx. $2.1\% \Delta V/V/\text{sub f}/\text{per } 10^{22}$ n/cm². Neutron irradiation of Type-321 stainless steel to a fluence level of 9.0×10^{22} n/cm², $E > 0.1$ MeV, shows swelling similar to that for Type-304 with a peak swelling temperature in the vicinity of 450°C. Cold working is more effective in reducing swelling in Type-316 than in Type-304 or Type-321.

1233 (GEFR--00067-2) PROGRAM FOR THE DEVELOPMENT OF DESIGN DATA: LMFBR STEAM GENERATOR MATERIALS. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). 7 Jun 1978. Contract EY-76-C-03-0893-030. 241p. AT.

A program to characterize materials, mainly 2 1/4 Cr-1 Mo steel and austenitic stainless steels, to assure that satisfactory materials compatibility is achieved with LMFBR steam generator environments is reported, along with a program for steam generator materials qualification. The objectives of the second program are to develop and qualify materials specifications, to develop fabrication procedures and processes, to purchase tubing and tubesheet material for development programs, to develop and apply nondestructive examination techniques for the Intermediate Heat Transport System of the CRBRP, to coordinate design of the steam generators in the area of water chemistry, to provide technical management to wear and friction testing of IHTS material, and to develop the design and fabrication procedures for the austenitic stainless to ferritic steel

transition joints of the CRBR. (GHT)

- 1234 (GEFR--00067-3) PROGRAM FOR THE DEVELOPMENT OF DESIGN DATA LMFBR STEAM GENERATOR MATERIALS. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jun 1979. Contract EY-76-C-03-0893-030. 140p. AT.

Progress in steam generator materials development and qualification for LMFBRs and the Clinch River Breeder Reactor is reported.

- 1235 (GEFR--00076) REFERENCE FUEL STUDIES. ANNUAL REPORT, FEBRUARY 1976--JANUARY 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Feb 1977. Contract EY-76-C-03-0893-003. 236p. AT.

Research progress is reported for the following subtasks: fuel pin testing and analysis; in-reactor behavior of reference structural materials; and fuel pin chemistry and thermodynamics.

- 1236 (GEFR--00101) NATIONAL CORE RESTRAINT PROGRAM CODE VERIFICATION/COMPARISON STUDY, PROBLEM LEVEL I=D. Fox, J.N. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Apr 1977. Contract EY-76-C-03-0893-023. 37p. AT.

The CORTAC computer code calculates the short term steady-state and transient response of the core restraint system. This requires the simultaneous analysis of the coupled effects of reactor core thermal-hydraulics, reactor kinetics, and component structural deflections. The structural portion of CORTAC is being verified along with the other core restraint codes PARABOW, NUBOW-2D and NUBOW-3D, under Problem Levels I-A, I-C, IV-A, and IV-B. However, CORTAC is the only core restraint code that treats reactor thermal hydraulics and reactor kinetics. Thus, the Level I=D Problems are unique in that they only apply to CORTAC. The Level I=D Problems are directed primarily to the reactor kinetics related and the transient portions of CORTAC. The CORTAC core-side steady-state radial heat transfer model will be verified under Problem Levels I-B and II-B, in which the CORTAC results will be compared directly to the WARD TRITON code which rigorously treats 3-D steady-state heat transfer. There are three types of problems under Level I=D: feedback reactivity, kinetics response, and transient heat transfer response. Each of these problems is sufficiently important so that it is essential to examine the results individually without the influence of any other effects. The report describes the feedback reactivity problem in which the inlet temperature is increased by 10°F at a zero power condition and hand calculations of the CORTAC results. The description and results of the point kinetics equations are provided with and without reactivity feedback. Each kinetics problem is subject to a step change in input reactivity. Two transient heat transfer response problems are considered: axial convection and fuel-to-coolant radial heat transfer.

- 1237 (GEFR--00108) ANALYSIS OF THE ZPPR-4 CRITICAL EXPERIMENTS, PHASES 3 AND 4. Hartman, A.K.; Hitchcock, J.T. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Aug 1977. Contract EY-76-C-03-0893-005. 149p. AT.

Assembly 4 of ZPPR was a systematic study of the nuclear performance of a demonstration plant sized LMFBR. Its primary purpose was to provide a benchmark study of the radial

blanket. Specifically, the objective of ZPPR-4 was to investigate plutonium buildup and breeding performance, and to provide a data base on the family of effects related to fuel burnup. The program was also used to provide pre-Engineering Mockup Critical data to be applied to the Clinch River Breeder Reactor design. The ZPPR-4 program was divided into four experimental phases in order to investigate the separate and combined influence of control and plutonium buildup. The purpose of Phases 3 and 4 was to investigate the neutronic characteristics of a simulated equilibrium core and to study the effects of blanket fissile buildup on core and blanket performance. The analysis of the ZPPR-4 measurements was directed toward the evaluation and calibration of standard design techniques as they are applied in LMFBR design. An attempt was made to lower and quantify the nuclear design uncertainties. Where possible, the standard design methods were calibrated against the experimental data. Where normal design methods failed or were inadequate. The necessary analytical effort to adequately analyze the experimental data was identified. The primary calculational tool was few group two-dimensional diffusion theory. The cross section data base was ENDF/B Version III.

- 1238 (GEFR--00109) ANALYSIS OF THE ZPPR-5 CRITICAL EXPERIMENTS: SIMULATING HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS. Lee, B.W.; Crick, S.C. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jul 1977. Contract EY-76-C-03-095. 172p. AT.

An experimental program was performed in the ZPPR critical facility in order to evaluate the design methods used to provide the nuclear feedback information for the hypothetical core disruptive accident analysis. A series of "snapshot" views of a progressing accident was mocked up in the reactor and measurements were made of the resulting reactivity. The accident scenario consisted of voiding sodium, followed by slumping 30 percent of the steel, followed by slumping fuel both toward and away from the core center. The ZPPR-5 measurement program has been broken into two phases, A and B. In Phase A, control rods were parked in the upper axial blanket and sodium-filled control rod channels were located in the core and lower axial blanket to simulate an initial core at end-of-cycle. Phase B, simulated beginning-of-cycle condition, was created from Phase A by inserting seven control rods from parked position in the upper axial blanket into the core region. The analytical emphasis was on the evaluation and calibration of standard design techniques. An attempt was made to lower and quantify the nuclear design uncertainties. The primary calculational tool was twenty-one group, two-dimensional diffusion theory in R,Z geometry. Both perturbation worths and eigenvalue differences were generated and compared with measurements. The cross section data base was ENDF/B Version III.

- 1239 (GEFR--00114(L)) EVALUATION OF HYDRAULIC TEST MODEL (HTM) FLOW DISTRIBUTION TEST RESULTS. Lee, Y.J. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). May 1977. 37p. AT.

The CRBR steam generator Hydraulic Test Model (HTM) flow distribution test results have been evaluated, and it is found that the performance degradation due to the flow nonuniformity observed in the HTM test is negligible; 0.06% for the evaporator and nil for the superheater. Also, the fact that ambient temperature water was used in the test instead of hot sodium with temperature

gradients does not introduce significant errors in applying the HTM test results for the tested flow rate range to the CRBR steam generator flow pattern.

- 1240 (GEFR--00152) PROCEDURES FOR MEASURING BOWING/EXPANSION REACTIVITY EFFECTS. Fox, J.N. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Nov 1977. Contract EY-76-C-03-0893-023. 31p. AT.

Detailed analyses of CRBRP beginning-of-cycle first core were performed with the computer code CORTAC. The objective was to provide a sound technical basis for development of suitable reactivity measurement procedures. The resulting recommended procedures provide for the separate measurement of core restraint reactivity-related quantities during reactor startup and rise-to-power, radial reactivity feedback based on a transient which suppresses the Doppler and axial feedback contributions, and Doppler plus axial reactivity feedbacks based on a transient designed to suppress the radial reactivity feedback. For the model used, it was found that the transient reactivity feedbacks of coolant density, structural density, and cladding density were negligible compared with the fuel-related and radial reactivity feedbacks. The coolant density reactivity feedback was found to be less than 2% of the total, the structural density contribution essentially zero, and the cladding density reactivity feedback less than 1% of the total. During the startup and rise-to-power phase, the sodium density contribution is a little higher (< 3%) while the structural and cladding density reactivities remain about the same as in the transient case. Hence, the only components of reactivity feedback that are important relative to the reactivity measuring procedures contained herein are Doppler, axial, and radial feedback. In addition to providing total reactivity feedback as a function of power, the recommended reactivity measurement procedure during the startup and rise-to-power sequence provides for two key core restraint quantities that can be implied through the total reactivity measurements. The first quantity is the onset of the structurally determinate region; the second is the power level at which the Above-Core Load Plane (ACLP) gaps are effectively closed.

- 1241 (GEFR--00156) CORE RESTRAINT SYSTEM SEISMIC MODELING CAPABILITIES. Fakhari, M.; Fox, J.N. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Nov 1977. Contract EY-76-C-03-0893-023. 249p. AT.

Due to the lack of adequate analytical tools and supporting test data, seismic response is the most difficult design area of core restraint. The primary concerns of seismic response are: the reactivity changes associated with both horizontal and vertical motions of the core assemblies; the structural integrity of the core assembly load pads and duct walls under the influence of seismic loading; and the horizontal displacements of the upper ends of the core assemblies which could potentially interfere with control rod insertion. The information herein addresses both analytical modeling and testing; however, this information is preliminary since it represents less than 20% of the overall effort required to adequately evaluate core restraint system seismic response.

- 1242 (GEFR--00232) INSTRUMENT DEVELOPMENT: ADVANCED SIGNAL TRANSMISSION SYSTEMS. FINAL SUMMARY REPORT, OCTOBER 1976-JUNE 1977.

Imhoff, D.H.; Craig, C.N. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Aug 1977. Contract EY-76-C-03-0893-009. 26p. AT.

The Instrumentation Development Program was initiated in January 1972 with the objective of developing in-core and in-vessel instrumentation for application to LMFRs. The objective of the subtask reported is to develop methods of monitoring reactor conditions and signal transmission to reduce in number and to simplify or eliminate vessel head penetrations for in-vessel instrumentation. Feasibility studies continued on methods of monitoring temperatures and transmitting acoustic energy via waveguides. Fluidic switching and sampling system work was completed. Feasibility of acoustic methods in application to subcriticality monitoring for the Clinch River Breeder Reactor was investigated and one technique found suitable for further development.

- 1243 (GEFR--00239) NATIONAL CORE RESTRAINT DEVELOPMENT PROGRAM CODE VERIFICATION/COMPARISON STUDY, PROBLEM LEVEL I-B. Fox, J.N. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Aug 1977. Contract EY-76-C-03-0893. 23p. AT.

The CORTAC computer code calculates the steady-state and short-term transient response of the core restraint system. This requires the simultaneous analysis of the coupled effects of reactor core thermal-hydraulics, reactor kinetics, and component structural deflections. The structural portion of CORTAC is being verified along with the other core restraint codes PARABOW, NUBOW-2D and NUBOW-3D, under Problem Levels I-A, I-C, IV-A, and IV-B. However, CORTAC is the only core restraint code that treats reactor kinetics and thermal-hydraulics. Thus, the verification problems associated with reactor kinetics (Problem Level I-D) and thermal-hydraulics (Problem Levels I-B and II-B) are unique in that they only apply to CORTAC. The Level I-D Problems, already completed, were directed primarily to the reactor kinetics-related and transient portions of CORTAC. The CORTAC core-wide steady-state radial heat transfer model is verified under Problem Levels I-B and II-B in which the duct wall temperatures are compared directly to the WARD TRITON code results which rigorously treat 3-D steady-state heat transfer. The Level I-B Problem, the subject of the document, is the verification of CORTAC for a seven-assembly cluster.

- 1244 (GEFR--00255) LMFR REFUELING FORCE LIMITS CRITERIA, TESTING AND EVALUATION. Fox, J.N. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1977. Contract EY-76-C-03-0893-031. 54p. AT.

This report develops criteria for refueling force limits by means of analysis and testing. These criteria are applied to a CRBRP refueling model to assess the practicality of the criteria and to provide an example application of their usage. It was determined that refueling force limits should be based on not exceeding core component damaging lateral loads with the axial structural limits and the refueling equipment designed to meet or exceed the corresponding refueling force limits. A damaging lateral load is defined as a lateral load that would result in load pad galling, duct wall galling, transfer of metal to the nosepiece liner or permanent distortion of the receptacle liner. The corresponding lateral load limits have been established for prototypical materials and geometries.

- 1245 (GEFR--00256) PECS-III: PROBABILISTIC EVALUATION OF CLADDING LIFETIME IN LMFBR FUEL PINS. Lovejoy, W.S.; Patel, M.R.; Hoover, D.G.; Krommenhock, F.J. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Oct 1977. Contract EY-76-C-03-0893-011. 220p. AT.

Theoretical basis, user instructions, a sample whole-core analysis and program listing for the probabilistic FBR fuel pin structural analysis code PECS-III (Probabilistic Evaluation of Cladding Lifetime) are described. Monte Carlo simulation is used to generate probability distributions for time to failure, cumulative damage fraction and other significant parameters in fuel pin cladding subjected to the fission gas pressure and fuel-cladding mechanical interaction loads under normal operating conditions. Importance sampling is used to reduce the number of Monte Carlo samples required for adequate estimate of the low probability of failures. A multiple regression analysis capability in the code permits correlation between selected input parameters and PECS-generated response parameters. The code is operational on the Honeywell-6000 and CDC-7600 computers.

- 1246 (GEFR--00257) INHALATION DOSE CONVERSION FACTORS FOR VARIOUS RADIONUCLIDES ASSOCIATED WITH LIQUID METAL FAST BREEDER REACTORS. PART 2. Teresi, J.D. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1977. Contract EY-76-C-03-0893-015. 27p. AT.

This report presents the values of the inhalation dose conversion factors (F-factors) in Rem per curie inhaled for 30 important LMFBR radionuclides. In addition, an example of a parametric analysis is presented in the calculation of ^{90}Mo F-factors for seven particle sizes and three clearance classes for each particle size.

- 1247 (GEFR--00258) REFERENCE FUEL STUDIES SEMI-ANNUAL REPORT, FEBRUARY--JULY 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Oct 1977. Contract EY-76-C-03-0893-003. 128p. AT.

Research progress is reported for the following LMFBR-related subtasks: (1) fuel pin chemistry and thermodynamics, and (2) fuel pin testing and analysis.

- 1248 (GEFR--00260) PROTOTYPE LARGE BREEDER REACTOR. PHASE II. EXTENSION STUDIES. PART 2. REACTOR ASSEMBLY. Davies, S.M. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1977. Contract EY-76-C-03-1143. 206p. AT.

Information is presented concerning the seismic analysis of the PLBR loop design; limiting temperature conditions for the reactor vessel roof structures; reactor vessel support and roof thermal/structural analysis; and reactor vessel closure (rotating plugs).

- 1249 (GEFR--00260) PROTOTYPE LARGE BREEDER REACTOR. PHASE II. EXTENSION STUDIES. PART II. REACTOR ASSEMBLY. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1977. Contract EY-76-C-03-1143. 203p. TIC.

Portions of document are illegible. The seismic analysis of the PLBR loop reactor design was accomplished with two distinct finite element models in order to obtain a complete picture of the stresses in the reactor assembly. Although only one model

is discussed in detail, this summary discusses the results of both the models. The axisymmetric model and the stick model were subjected to the same input seismic excitation. Two conservative approximations of the complete PLBR response spectrum were used. They were the spectra for horizontal excitation specified by: (1) the USAEC Regulatory Guide 1.60 and (2) the Bechtel Corporation based on a preliminary analysis of the PLBR structure. Two-thirds of each of the horizontal spectra were applied to the models in the vertical direction in order to investigate the reactor response to vertical motion.

- 1250 (GEFR--00260(Pt.1)(Summ.)) PROTOTYPE LARGE BREEDER REACTOR. PHASE II EXTENSION STUDIES. PART I. SUMMARY. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1977. Contract EY-76-C-03-1143. 24p. AT.

The Prototype Large Breeder Reactor design project has been performed in two distinct work phases. During the Phase I the plant characteristics and design criteria were established during a period of tradeoff studies. The major features established during Phase I were: power, temperature, steam cycle, number of loops, component and system features, refueling schemes, and research and development needs. During Phase II an overall conceptual design for PLBR was established and the initial plant design essentially completed in the following areas: core, components, systems, structures, research and development needs, and project schedule. The reactor assembly and heat transport systems are described.

- 1251 (GEFR--00260(Pt.3)) PROTOTYPE LARGE BREEDER REACTOR. PHASE II. EXTENSION STUDIES. PART III. HEAT TRANSPORT SYSTEMS. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1977. Contract EY-76-C-03-1143. 360p. AT.

The transient analysis during PLBR Phase II Extension was broken into four specific areas. These are (1) to determine the sensitivity of the reactor scram transient for both the saturated and superheat cycle designs to a range of possible decay heat and initial power levels, (2) to investigate the potential for improving the reactor scram transient for the superheat cycle design (3) to evaluate selected transients during single-loop component failures without a reactor scram for the saturated cycle design and (4) to evaluate the drum vent failure transient followed by a reactor scram for the saturated cycle design.

- 1252 (GEFR--00260(Summ.)) PROTOTYPE LARGE BREEDER REACTOR. PHASE II EXTENSION STUDIES. PART 1. SUMMARY. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1977. Contract EY-76-C-03-1143. 24p. AT.

Information is presented concerning the reactor vessel seismic analysis, roof structure, support, and closure; and heat transport systems including steam generator, sodium pump design, and plant transient analysis.

- 1253 (GEFR--00274(L)) PLUGGING/UNPLUGGING OF DEFECTS IN LMFBR STEAM GENERATOR TUBING. Fellow, H.C. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1977. Contract EY-76-C-03-0893-010. 30p. AT.

An investigation was made to (1) determine the plugging tendency of small leaks/defects in

steam generator tubing material when exposed to simulated manufacturing and testing techniques, and (2) determine the ability of a candidate plant cleaning solution to dissolve the plug and open the leak. Nine series of four leaks each (leak sizes ranging in equivalent steam flow rate from 1.34×10^{-5} to 1.2×10^{-1} grams/sec) were exposed to: humid storage with VPI addition, die lubricant, heat treatment, hydrotest, dye penetrant examination, sodium exposure, centerless grinding lubricant and sodium-water reaction products. Severe reduction in leak flow in almost all cases resulted from exposure to the various processes, with complete plugging of about $\frac{1}{3}$ of the leaks. Cleaning of the defects with a 4% citric acid solution containing 0.1% Rodine 33, at a pH of 3.0 to 4.0 and a temperature of 160°F (71°C) resulted in the opening of all the leaks tested (including those completely plugged) to the same order of magnitude as the large leaks in the series. The use of such a cleaning solution following tube fabrication, before operation of a steam generator, and/or after a leak has been detected but subsequently plugged, appears to be feasible. Additional work in this area is recommended to provide the operational techniques required.

- 1254 (GEFR--00279(L)) PRELIMINARY SELF WASTAGE TEST RESULTS FROM CRACKPOT FACILITY. Gudahl, J.A.; Hunsicker, J.C.; Gerrels, E.E. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1977. Contract EY-76-C-03-0893-010. 38p. AT.

This preliminary report presents results of the microdefect self-wastage tests conducted in the new CRACKPOT facilities as of September 1977. Twelve tests were completed; three additional tests are in progress. Complete self-wastage penetration of the tube wall (e.g., defect enlargement) occurred in three cases at 900°F (482°C), three at 650°F (343°C), and one at 580°F (304°C). In three tests at 580°F and one test at 650°F, the leak rate remained below the initial rate during the entire test. Complete penetration was not achieved in one test at 650°F prior to test termination. All three of the continuing microdefect tests, at 650°F, are in a plugged condition.

- 1255 (GEFR--00291) VENTED ASSEMBLY CORE DEVELOPMENT PROGRAM, FY-77. Dubberley, A.E. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1977. Contract EY-76-C-03-0893-020. 186p. AT.

Information on LMFBR vented assembly cores is presented concerning thermal-hydraulic performance, vent flow uncertainties, load pad mechanical performance, and core economic performance.

- 1256 (GEFR--00293) SUMMARY OF THE CORTAC VERIFICATION ACTIVITY AND THE EVALUATION OF 3-D EFFECTS. Fox, J.N.; Wei, J.P. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1977. 23p. AT.

The report documents the results of the FY-77 Core Restraint Transient Analysis Subtask C effort. The objectives of this subtask are to complete the verification of the CORTAC computer code and to perform an evaluation of 3-D effects relative to the current capability of the code. The CORTAC computer code calculates the short-term steady-state and transient response of the core restraint system. This requires the simultaneous analysis of the coupled effects of reactor core thermal-hydraulics, reactor kinetics, and component structural deflections. The structural portion of CORTAC is being verified along with the

other core restraint codes NUBOW-2D, PARABOW, and NUBOW-3D, under verification problem levels I-A, I-C, IV-A, and IV-B. However, CORTAC is the only core restraint code that treats reactor thermal-hydraulics and reactor kinetics. Thus, the level I-D problem is unique in that it only applies to CORTAC. The level I-D problem is directed primarily to the reactor kinetics related and the transient portions of CORTAC. The CORTAC core under steady-state radial heat transfer model is verified under problem levels I-B and II-B, in which the CORTAC results are compared directly to the WARD TRITON code which rigorously treats 3-D steady-state heat transfer. The results of this verification study demonstrated the need to develop 3-D models for the eventual inclusion into a 3-D version of CORTAC. The report summarizes the CORTAC verification activity, identifies essential 3-D effects, and provides a CORTAC-3D development plan.

- 1257 (GEFR--00295) DATA-BASED EVALUATION OF THERMAL PERFORMANCE MECHANISMS IN MIXED-OXIDE FUEL PINS. Vaidyanathan, S.; McCarthy, W.H.; Hilbert, R.F. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1977. Contract EY-76-C-03-0893-031. 75p. AT.

Evaluations of data from GE F-20 experiment indicate that the thermal performance of mixed oxide fuel can be broadly classified into two regimes: early life thermal performance and high burnup thermal performance. During early life operation (less than 1% burnup) the power to incipient melt of fresh fuel is greatly increased by the processes of fuel restructuring and fuel cladding gap closure. The power to melt of fresh fuel immediately after startup is in the range of 16--17 kW/ft. Subsequent to restructuring and gap closure processes which happen within 1% burnup provided the power level is sufficiently high, the power to melt increases to 20--22 kW/ft. If, however, the operating power levels are low (less than approximately 10 kW/ft) this increase does not take place. At high burnup (>5%), the F-20 data indicate a loss of thermal performance. Pins that operated at high power levels and underwent early life restructuring values exhibit incipient power to melt values in the 15--16 kW/ft range at high burnup. The power to melt of pins that operated at low powers and which did not undergo early life restructuring is decreased to 12.5--14 kW/ft. This decrease in power to melt is thought to arise as a result of decreased fuel thermal conductivity at high burnup. Thermal performance maps have been generated for each of these regimes. Based on preliminary evaluations that suggest the fuel conductivity at high burnup to be a function of initial fuel density, it is recommended that pins be designed with the highest practical fuel density. With the aid of the high burnup thermal performance map, confirming evidence for the proposed models has been obtained from experimental irradiations that were not specifically designed to be power to melt experiments.

- 1258 (GEFR--00303) FATIGUE CRACK GROWTH PROPERTIES OF TYPE-316 PLATE AND 16-8-2 STAINLESS STEEL WELDS IN AIR AND HIGH CARBON LIQUID SODIUM. Copeland, J.F.; Yuen, J.L. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1977. Contract EY-76-C-03-0893-031. 119p. TIC \$6.50.

The fatigue crack growth rates of Type 316 stainless steel and 16-8-2 GTA welds were measured in air and high-carbon liquid sodium

at 513°C (950°F) for secondary sodium loop LMFBR applications. Loading variables such as low frequency, load ratio, and hold-time at peak load were studied. Crevice experiments were performed, and a diffusion model for carburization in a crack was developed. The results show that carburization of crack faces occurs prior to any crack tip carburization for up to 5 to 15% of a 2.1×10^5 hr. service life, with practical crack sizes and load frequencies. Thus, a carburization effect on da/dN is not shown for the relatively short-term tests of this study. Low AK, ultra-low frequency tests to simulate service would be required to show this effect. These tests appear to be very time consuming. It is recommended that Nuclear Systems Materials Handbook (NSMH) crack growth curves for austenitic stainless steels in air, with appropriate frequency and load ratio corrections, should be conservative for design against crack extension in high-carbon sodium, for base metal and welds. Verification of this recommendation by possible ultra-low frequency testing and long-time exposure of surveillance crevice specimens under service conditions is also recommended.

- 1259 (GEFR--00305) IRRADIATION AND POST-IRRADIATION EXAMINATION DATA FROM SELECTED F9 AND F5 TEST PINS. Gatus, L.A.; Hilbert, R.F. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Nov 1977. Contract EY-76-C-03-0893-031. 121p. AT.

This document presents the design, operating parameters, and post-irradiation examination results from four mixed oxide fuel pins from the GE F9 and F5 test series that were irradiated in EBR-II. Examination of pins from the F9 series, which cover burnups up to 19 atom %, continuing at LASL. This report covers the F9 and F5 pins for which post-irradiation data became available for reporting during FY 1977.

- 1260 (GEFR--00320(Vol.1)) CLINCH RIVER RELOAD CORE DESIGN STUDY. VOLUME I. TRADE-OFF STUDY AND DESIGN RECOMMENDATIONS. Deane, N.A. (comp.). (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Feb 1978. Contract EY-76-C-03-0893-031. 252p. AT.

This is the final report of the General Electric portion of a year-long study to conceptually design advanced reload replacement cores for the CRBR. General Electric was to examine the potential of advanced oxide fuel replacement cores using advanced design concepts and advanced materials. Two objectives were established at the outset for the study. They were: to study the potential for improvements in compound system doubling time (CSDT) for future core reloads; and to study the potential for increases in thermal power output for future core reloads. General Electric analyzed in detail a minimum CSDT core and a high power (20 to 30 percent greater than the current design) core in order to understand the total performance potential for CRBR. A 40-year CSDT core which represents a 15-year improvement over the reference core was also analyzed during the trade-off studies.

- 1261 (GEFR--00320(vol.2)) CLINCH RIVER RELOAD CORE DESIGN STUDY. VOLUME II: RECOMMENDED CORE DESIGN EVALUATION. Deane, N.A. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Feb 1978. Contract EY-76-C-03-0893-031. 334p. \$12.00.
Information on the Clinch River Breeder Reactor reload core design is presented

concerning neutronic performance evaluations, recommended core thermal-hydraulic performance, mechanical performance, CRBR constraints, and economic impact of the recommended core designs.

- 1262 (GEFR--00321) TERMINAL EXAMINATION OF THE F9E AND F9F EBR-II GRID-SPACED FUEL ASSEMBLIES. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Feb 1978. Contract EY-76-C-03-0893-031. 71p. AT.

The F9E assembly of 37 pins was loaded into EBR-II in July 1972 and completed its burnup after 17,000 hours of reactor operation in February 1975. The F9E experiment achieved a burnup of 11.8 atom % peak in the lead pin. The F9E experiment was designed to simulate high fluence mechanical interaction between fuel pin and grid spacer. The F9E accomplished its objective by using annealed Type-316 stainless steel rather than cold-worked stainless steel for fabrication of the duct, fuel pin cladding, and grid spacers. This allowed the experiment to obtain high relative pin-to-grid swelling on a shortened low fluence time scale. The F9E fuel pins were supported by two types of grid spacers: (a) the main support of six "full" grids at 6-inch intervals and (b) three "staggered" grids at 3-inch intervals. Three months after insertion of the F9E experiment, in October 1972, the 37-pin F9F experiment was placed in EBR-II. The F9F experiment achieved a burnup of 12 atom % in the lead pin after approximately 20,000 hours of EBR-II operation. The F9F experiment was designed to simulate high fluence mechanical interaction between the fuel pin bundle and the duct. To achieve the desired bundle-to-duct interaction, an increased spacing was used between the edge fuel pins and the duct wall. The increased edge distance resulted in an overcooled duct relative to the fuel pins and grids. In this way, high beam-to-duct and pin-to-duct relative growths were obtained. The structural material used for the fuel pins, grid spacers, and the duct was 20% cold-worked Type-316 stainless steel. The staggered grid concept was used almost exclusively in the F9F experiment but for two full grids and a single tabless grid placed near the top of the fuel pins. The success of these experiments provide a high level of confidence that the planned grid-spaced experiments in FFTF will be successful, and that burnups in excess of 100,000 MWD/Te are achievable.

- 1263 (GEFR--00322) FRICTION, WEAR, AND FRETTING: CLINCH RIVER BREEDER REACTOR STEAM GENERATORS. Hartle, R.T.; Ring, P.J. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Apr 1978. Contract EY-76-C-03-0893-030. 66p. AT.

Steam generator tubing is subject to fretting, wear and galling due to movement of the tubing relative to the spacer plates. Testing was performed to determine the degree of wear likely for 2 1/4%Cr-1Mo under Clinch River Breeder Reactor Steam Generator conditions.

- 1264 (GEFR--00324) SOURCE TERM MODEL NEEDS AND DEVELOPMENTS. McDonald, J.W.; Huang, T.C.; Kriebel, A.R.; Stitt, R.K.; Teresi, J.D. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Feb 1978. Contract EY-76-C-03-0893-032. 47p. AT.

This report summarizes source-term model developments from the beginning of this base program, Radiological Assessment, through September 1977. The summary is intended to show

the scope, status, and needs of radiological source-term technology. The approach to model development has been to divide HCDA accident scenarios into phases and to work on those phases which offer the most potential benefit. The work covers energetic and melt-through HCDA scenarios, and is directed to demonstrating for extremely low probability accidents that public safety is assured.

- 1265 (GEFR--00340) ACOUSTIC LEAK DETECTION SYSTEM CONCEPT DESIGN. Greene, D.A. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jan 1978. Contract EY-76-C-03-0893-030. 101p. AT.

Design criteria are developed for acoustic monitoring of small sodium-water reactions in an LMFBR steam generator. From these criteria a concept design of an acoustic leak detection system has been prepared. The system description includes discussions of predicted system costs, expected system performance, mode of operation, and reactor manufacturers target cost criteria for leak protection systems. It is concluded that an acoustic system is feasible for LMFBR steam generators.

- 1266 (GEFR--00347) UPDATED ANALYSIS OF THE ZPPR-7 CRITICAL EXPERIMENTS: PHASES A, B, AND C. Hartman, A.K.; Kujawski, E. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Feb 1978. Contract EY-76-C-03-0789-031. 87p. AT.

The ZPPR-7 experimental program involved a study of several tightly coupled heterogeneous cores at various points in life. One of the objectives of the program was to evaluate the accuracy of analytical methods for predicting the nuclear behavior of heterogeneous cores. In the present work a number of measurements taken in Phases A, B, and C were reanalyzed using ENDF/B-IV data, and the emphasis was on investigating the sensitivity of the predictions to the nuclear data, data processing and neutronics methods. Calculations for criticality, neutron reaction rate distributions, control rod worth, and sodium void worth are reported. Changes in data and data processing were found to be important and to impact differently on homogeneous and heterogeneous cores. Transport corrections are important for criticality and control rod worth predictions in heterogeneous cores. Mesh effects are especially significant for control rod worth calculated with diffusion theory. Criticality is generally underpredicted by 1.0 to 1.5%. Transport corrections for Phase B are 0.43% $\Delta k/k$. The average (C/E) for the control rod worth in Phases B and C using four meshes per drawer and transport theory is 0.96, while using one mesh per drawer and diffusion theory the average (C/E) for the control worth is 0.91. The present analysis indicates the need for accurate cross sections and refined neutronics methods when analyzing heterogeneous cores. The general conclusion is that the nuclear performance of heterogeneous cores can be predicted with an accuracy consistent with homogeneous cores when due care is given to the analysis. The calculated parameters for Phase C, however, are in better agreement with the measurements than those for Phase B, which is indicative of the greater difficulties in accurately predicting heterogeneous cores. Remaining discrepancies appear to be consistent with the uncertainty in the nuclear data.

- 1267 (GEFR--00348) HEAT TREATMENT FOR 2-1/4Cr-1Mo FOR APPLICATION TO SODIUM HEATED STEAM GENERATORS. Spalaris, C.N.; Roy, P.; Ring, P.J.; Pizzo, P.P.; Hampton, L.V. (General

Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Feb 1978. Contract EY-76-C-03-0893-031. 17p. AT.

The reason is summarized for selecting the isothermal and full anneal heat treatments for 2-1/4Cr-1Mo. Supportive data which have been accumulating as a result of DOE-sponsored base development programs, foreign experiments, as well as recent wrought product fabrication by commercial vendors, indicate that decisions made in mid-1974 regarding heat treatment selection were justified.

- 1268 (GEFR--00349) HYDRAULIC TESTING PROGRAM ON VENTED ASSEMBLY CORE DESIGN. Wang, J.C.F. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Mar 1978. Contract EY-76-C-03-0893-031. 73p. AT.

Early LMFBR vented core hydraulic studies demonstrated that vent and interstitial flows are primarily governed by the vent and load pad plane flow resistances. The methods used to evaluate the magnitude of these resistances were very simplified and thus lead to large uncertainties in the vent and interstitial flow patterns. To reduce the largest of these uncertainties, an analytical and test program was initiated in FY-77 to provide better flow resistance correlations with empirical verifications for (1) flow through vents, (2) flow past load pads, and (3) radial flow between assemblies. This program was completed by the end of FY-77 and the report completes the documentation.

- 1269 (GEFR--00351) LOCALIZED DUCT WALL CYCLIC THERMAL STRESSES IN VENTED ASSEMBLY LMFBR CORES. Dubberley, A.E. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Dec 1977. Contract EY-76-C-03-0893-031. 85p. AT.

A vented assembly differs from a standard assembly only by having one or more vents machined into the duct wall at positions above the active core. The vents below the Above Core Load Pads (ACLP's) allow coolant to leave the assembly and pressurize the interstitial spaces between the assemblies before flowing up past the load pads and exiting to the upper plenum. Additional vents above the ACLP's further reduce the assembly pressure drop. Two primary benefits are anticipated from a core composed of vented assemblies. First, a pressurized interstitial space reduces the pressure across the assembly duct wall, reducing duct stress and deflection and allowing either longer dilation life or a thinner duct wall. The second benefit is a reduced pressure drop between the reactor inlet and outlet plenum for the same core coolant mass flow rate. The flow exiting from the assemblies at the vents and following a "parallel" hydraulic path through the interstitial region to the outlet plenum reduces the mass flow rate and thus the pressure drop in the upper portion of the assemblies. The greater the bypass flow leaving the vents, the greater the reduction in assembly pressure drop will be.

- 1270 (GEFR--00352) MICROLEAK WASTAGE TEST RESULTS. Gudahl, J.A.; Magee, P.M. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Mar 1978. Contract EY-76-C-03-0893-030. 83p. AT.

The report presents results of, and conclusions drawn from, the microdefect self-wastage tests conducted on 2-1/4Cr-1Mo tubing samples in the new sodium filled Crackpot facilities. A total of fifteen sodium-water reaction tests were run at three temperature levels: 580°F (304°C), 650°F (343°C) and 900°F

(428°C). In seven tests, complete self-wastage penetration of the tube wall occurred under constant temperature conditions. In three tests at 580°F and one at 650°F, the leak rate remained below the initial leak rate during the entire test. Four tests at 650°F plugged completely and remained plugged for the duration of the tests, in three cases for more than two months. These three plugged specimens were then thermally cycled; only one reopened. All steam leaks (650°F and 900°F) initially less than 10^{-5} lb/sec (4.5×10^{-3} gm/sec) completely plugged for at least some period of testing. Conversely, only one leak initially greater than 10^{-5} lb/sec decreased as much as one decade in leak rate. Water leaks (580°F) showed less tendency to plug and, if plugging occurred, greater tendency to reopen than the steam leaks, although the latter were at higher temperatures. It was determined that the enlargement growth rate is slow enough at 580 to 650°F that the CRBRP evaporator tubesheet regions are fully protected against microdefects initially so small as to be undetectable by the chemical leak detection system. However, because self-wastage damage rates are considerably faster at higher temperatures, there is a significantly probability that growth of an initially undetectable microleak in a CRBRP superheater will result in impingement wastage damage and possible failure of adjacent tubes before leak detection and unit shutdown.

1271 (GEFR--00353) CHARACTERIZATION OF CRBRP STEAM GENERATOR SHELL PLATE. Offer, H.P.; Ring, P.J. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Mar 1978. Contract EY-76-C-03-0893 030. 58p. AT.

A section of CRBR steam generator shell plate was evaluated in the fully annealed and annealed plus post weld heat treatment (PWHT) conditions. The microstructural features were determined to provide a more complete understanding of the influence of the structure on the mechanical properties which were subsequently characterized. The results of these mechanical property tests have been compared to the mechanical property guidelines used for design purposes, which in most cases were based on data from material with nonprototypic anneals and without PWHT. The results of this evaluation are presented.

1272 (GEFR--00354) COMPARISON OF SEMI-STATISTICAL HOT CHANNEL FACTOR AND MONTE CARLO ANALYSES OF FUEL ASSEMBLY COOLANT TEMPERATURE UNCERTAINTIES. Chen, K.H. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jun 1978. Contract EY-76-C-03-0893 034. 52p. AT.

A comparison between the semi-statistical hot channel factor and Monte Carlo analyses of the effects of uncertainties on the prediction of coolant temperature in a typical LMFBR fuel assembly was carried out under equivalent start of life design conditions. The design data base for major random and direct uncertainty parameters relevant to a thermal hydraulic analysis of an LMFBR fuel assembly was established from available reports and documents. The random, systematic and combined uncertainty effects were analytically and numerically investigated.

1273 (GEFR--00355(Rev.1)) BACKGROUND NOISE MEASUREMENTS ON PROTOTYPE OF SNR-300 STEAM GENERATOR IN THE 50 MW TEST FACILITY AT HENGEL, THE NETHERLANDS. Greene, D.A.; Ahlgren, F.F. (General Electric Co.,

Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Nov 1978. Contract EY-76-C-03-0893-030. 179p. AT.

Acoustic noise amplitudes were measured in a 50 MW helical coil prototype of a steam generator mounted in the Hengelo test facility in a joint technical program between GE/DOE and TNO/NERATOOM. Background acoustic noise measurements were completed with test conditions ranging from quiescent (with no sodium or water flow) to high thermal power transfer from the sodium circuit to the water circuit. The acoustic noise pressure on the vessel wall was measured using externally mounted accelerometers. Data were obtained for the effect of sodium flow rate on the acoustic noise amplitude. A preliminary correlation is derived for the acoustic noise generated by nucleate boiling of the water. Results indicate that acoustic leak detection/location is feasible for LMFBR steam generators.

1274 (GEFR--00357) CORE RESTRAINT SEISMIC TEST PLAN. Yang, T.M. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Mar 1978. Contract EY-76-C-03-0893-031. 18p. AT.

This test plan responds to Task 5 (Seismic Testing Plan) of the GE-FBRD plan for the evaluation of core restraint seismic response. The test plan has been developed with the assumption that the testing can be performed utilizing the FFTF Hydraulic Core Mockup (HCM) located at HEDL. The level of detail contained in this document is sufficient for HEDL to make cost and schedule estimates for the testing. This document also serves as a vehicle to obtain comments on the proposed seismic tests from the core restraint/seismic communities.

1275 (GEFR--00361) DEVELOPMENT OF SODIUM-WATER REACTION PRESSURE RELEASE RUPTURE DISCS FOR THE CLINCH RIVER BREEDER REACTOR PLANT. Reynolds, R.G. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jul 1978. Contract EY-76-C-03-0893-030. 91p. AT.

The report documents the requirements for the development of the CRBRP main duplex rupture disc assemblies, to describe how design and testing of the assemblies met the requirements and to list those future activities necessary to completely fulfill the requirements. Included in the report is a description of the rupture disc manufacturing process. Results from single rupture disc tests run to data are interpreted to estimate the performance of the duplex discs to be used in CRBRP.

1276 (GEFR--00366) INTERIM EVALUATION OF DATA FROM SODIUM-WATER REACTION TESTS SWR-3, SWR-4, AND SWR-5 AND COMPARISON WITH TRANSWRAP ANALYSES. SERIES I: LARGE-LEAK TEST PROGRAM. Regimbal, J.J.; Bradfute, J.D.; Rauch, T.; Sane, J.D. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jun 1978. Contract EY-76-C-03-0893-030. 159p. AT.

This evaluation treats the experimental results and analyses of performance of three additional sodium water reaction tests performed in the Large Leak Test Rig (LLTR) by ETEC (formerly LMEC). A series of six tests were conducted in the LLTR over the past few years. This report combines the evaluation of SWR-3, SWR-4, and SWR-5 in the continuing effort to provide a basis for validating the TRANSWRAP computer code. All three of the tests were DEG ruptures of a tube in the hockey stick region of the MSG with the same break location,

1.75 inches from the sodium face of the upper tube sheet. In SWR-3 the rupture tube was filled with a two phase mixture of steam and water; in SWR-4 the water side fluid was superheated steam and in SWR-5 nitrogen was used as the working fluid. The water side pressure was nominally 1900 psia for all three tests. These three tests, taken collectively, were intended to provide an experimental basis for an interpretation of the role of the pressurizing fluid vis-a-vis the products of the sodium/water reaction in producing the initial bubble pressure transient, since appreciably different flow rates of the liquid and vapor were delivered in SWR-3 and SWR-4 and non-reactive nitrogen was the pressurizing gas for SWR-5.

- 1277 (GEFR--00367) MICROSTRUCTURAL RESULTS FROM F20 POWER-TO-MELT EXPERIMENT. McCarthy, W.H.; Jedlovac, D.R. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). May 1978. Contract EY-76-C-02-0893-031. 172p. AT.

Data are provided on operating conditions, plenum gas, retained gas, and 113 microstructural specimens obtained from the GE-F20 power-to-melt experiment. The report extends the range of fuel and irradiation parameters that were contained in earlier reports on F20 Experiment results.

- 1278 (GEFR--00370) EXPERIMENTAL EVALUATION OF THE OXIDATION BEHAVIOR OF Nb, V, AND Ti AS CANDIDATE FCCI INHIBITORS. Caputi, R.W.; Adamson, M.G. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). May 1978. Contract EY-76-C-03-0893-031. 19p. AT.

Out-of-pile measurements of the rates of surface oxidation of the candidate cladding attack inhibitors Nb, V and Ti under typical operating fuel pin conditions, viz., low oxygen potential (Δ anti G/sub O₂/sup 750°C/ = -80 to -100 kcal/mol), temperatures in the range 600 to 900°C, presence and absence of low levels of cesium are described. The measurements were performed on clean metal foils in an automatic recording electrobalance system using flowing He-H₂-H₂O mixtures as the oxidizing atmosphere. To provide a basis for comparison, oxidation rate measurements were also performed on an annealed Type-316 stainless steel specimen. At 750°C the measured oxidation rates fall in the order V > Nb > Ti > 316 SS, but at 600°C the order is Nb > V > Ti. The linearized rates for Nb, V and Ti also compare favorably with values estimated from rates measured at higher oxygen pressures. The effect of Cs(v) vapor on the oxidation rates was studied. With p/sub Cs/ approximately 10⁻³ atms the catalytic effects proved to be relatively modest (30% increase for Ti, 70% increase for Nb, and 100% increase for V); however, as expected, Type 316 SS showed a dramatic increase in oxidation rate (13-fold increase). An attempt is made to explain the catalytic influence of cesium in terms of its reactivity with the various oxides of Nb, V, Ti and Cr.

- 1279 (GEFR--00373) INTERMEDIATE SYSTEM MOCKUP LOOP (ISML) OPERATION. Hampton, L.V.; Kanoh, S. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jun 1978. Contract EY-76-C-03-0893-031. 53p. AT.

The Intermediate heat Transport System (IHTS) of the Clinch River Breeder Reactor (CRBR) will experience carbon transport from the low alloy ferritic to the stainless steels. Loss of carbon will weaken the low alloy steel

and the resulting carburization may embrittle the stainless steel. The ISML test facility was constructed to model the carbon transport that will occur during the 30-year CRBR lifetime. The ISML simulates the materials, temperatures and surface area ratios of the CRBR. The objective of the tests performed is to provide the data base needed to design and predict the performance of components and piping associated with the CRBR Intermediate Heat Transport System. This report describes the ISML history to over 11,000 hours operating time.

- 1280 (GEFR--00375) CORROSION FATIGUE OF 2-1/4 CR--1 MO STEEL IN CAUSTIC AT 316°C. McIlree, A.R.; Pizzo, P.P.; Indig, M.E. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jun 1978. Contract EY-76-C-03-0893-031. 33p. AT.

The equipment developed to perform full reversed push-pull fatigue tests in a high temperature (316°C) aqueous environment (5% NaOH), and the effects of this environment on the fatigue life and failure mode of 2 1/4 Cr-1 Mo steel are described. The safety factor of the design allowable fatigue curve for 2 1/4 Cr-1 Mo steel adequately accounts for the possible degradation of properties in 5% NaOH at 316°C over the strain range investigated. The fracture morphology of 2 1/4 Cr-1 Mo steel at 316°C (500°F) is branching and transgranular in both air and 5% NaOH. Further high cycle (>10⁶) fatigue tests in 5% NaOH will be necessary to quantitatively establish the caustic influence. Equipment changes have been identified which will improve the strain measuring capability of the corrosion fatigue facility.

- 1281 (GEFR--00378) FINAL SUMMARY REPORT ON CHARACTERIZATION OF CRBRP STEAM GENERATOR TUBING. Offer, H.P.; Busboom, H.J.; Ring, P.J. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jun 1978. Contract EY-76-C-03-0893-030. 36p. AT.

A previous report (GEFR-00243-L) characterized the chemical composition, microstructural features, physical dimensions, and short term mechanical properties of a representative sampling of the initial lots of CRBRP steam generator tubing. This report summarizes these previous results, and gives the results of full wall thickness hydrostatic burst tests at room temperature. Also presented here are the results of reduced wall thickness stress-rupture tests at 510C (950F) in argon. The results of these additional tests are similar from lot to lot indicating (in agreement with previous testing) little variation in chemical composition and final heat treatment.

- 1282 (GEFR--00379) FINAL REPORT ON VACUUM ARC REMELTED TUBESHEET FORGING PROPERTIES FOR CRBRP STEAM GENERATORS. Offer, H.P.; Ring, P.J. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jul 1978. Contract EY-76-C-03-0893-030. 67p. AT.

The evaluation is presented of vacuum arc remelted (VAR) tubesheet material for CRBRP steam generators. The properties reported initially in General Electric Report GEFR-00245(L) are summarized. The additional work reported includes J-integral fracture toughness testing, tensile testing of tubesheet material reannealed by the tubesheet vendor, and creep-rupture testing results not previously available. Also reported are a limited number of tubesheet properties determined by the tubesheet vendor on material both before and after vendor reannealing.

- 1283 (GEFR--00384) TESTS AND ANALYSIS IN SUPPORT OF AN ACOUSTIC LEAK DETECTION/LOCATION SYSTEM (PROGRESS REPORT). Greene, D.A.; Gudahl, J.; Chronert, S.; Clark, E.J.; Ahlgren, F.F.; Scott, P.; Czechoski, J. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-030. 179p. AT.

General Electric, under Department of Energy sponsorship, has been developing a leak detection system based on detecting the acoustic noise produced by a small sodium/water reaction. The reaction noise is marked by background noise and is detected using an array of transducers. This interim progress report documents analytical and experimental investigations made in support of the detector development. A comprehensive investigation of array processing suggests that the current linear array approach is optimum for steam generator leak detection. Signal attenuation and delay between the source and transducer have been measured experimentally, and a computer code is available to predict the signal attenuation. Accelerometer choice and attachment techniques are outlined, with experimental validation. Basic electronic noise levels which can be achieved with commercial equipment have been measured.

- 1284 (GEFR--00386) VENTED ASSEMBLY CORE DEVELOPMENT PROGRAM, FY-78. Dubberley, A.E. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-031. 273p. AT.

The purpose of the Advanced LMFBR Core Design Program is to identify and evaluate core design trade-off issues for commercial-sized LMFBRs and to determine specific mechanical design options which lead to improved economic and nuclear performance. Scoping studies are used to define high priority advanced core design options. Detailed assessments of the high priority options covering economic, neutronic, thermal/hydraulic and mechanical performance then quantify attractive cost/benefit opportunities and identify significant potential problems. The objective of the FY-77, 78 and 79 vented core activities is the detailed assessment of vented core characteristics.

- 1285 (GEFR--00390) FUEL PIN FISSION GAS PLENUM LOCATION STUDY. Dubberley, A.E. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-031. 201p. AT.

The purpose of this study is to evaluate the interrelated effects of plenum design changes and the tradeoffs that are possible at the reactor system level: the reactor system economics, the core mechanics, and the reactor system safety. Reactor system economics are addressed by comparative evaluations of COROPT optimized 900 MWe loop LMFBR designs with upper plenum, lower plenum, and split plenum (both upper and lower, simultaneously). To evaluate the effects of advanced core design features on the plenum tradeoff issues, advanced (partial wireless) wire-wrapped bundles, staggered grid spaced bundles, vented assemblies and advanced material (D57) assemblies are also considered along with the reference, straight-start, wire-wrapped bundle in unvented assemblies. Brief descriptions of these advanced features are included.

- 1286 (GEFR--00392) PRECONCEPTUAL DESIGN STUDY OF PROLIFERATION-RESISTANT HOMOGENEOUS OXIDE LMFBR CORES. Taiwar, B. (General

Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Nov 1978. Contract EY-76-C-03-0893-031. 265p. AT.

The preconceptual design and performance of four 1000 MW(e) homogeneous oxide LMFBR cores are described. GE-ARSD developed the core designs as part of the Proliferation Resistance LMFBR Core Design Study (PRLCDS). The first design is the reference, using (PuU)O₂ fuel in the core and UO₂ in the axial and radial blankets. The second design is identical to the first, except that ThO₂ replaces UO₂ in the blankets. The third design is optimized for the use of (PuTh)O₂ in the core and ThO₂ in the blankets. Both of the second and third reactors produce the fissile isotope ²³³U and are referred to as transmuters. Either transmuter can form a symbiotic reactor system with the fourth reactor, which uses proliferation-resistant denatured (²³³U=²³⁸U)O₂ fuel with a fissile enrichment below 12%. The blankets of the denatured reactor contain ThO₂. All the designs were developed under a uniform set of ground rules. The performance evaluations consist of nuclear, thermal/hydraulic, mechanical, resource utilization, and preliminary safety evaluations. The PRLCDS thermal/hydraulic and mechanical performance cross-cutting study is also included.

- 1287 (GEFR--00393) ANALYSIS OF ADVANCED FUELS CRITICAL EXPERIMENTS. Zadworski, Z.F.; Indreboe, L.L. (General Electric Co., Sunnyvale, CA (USA)). Aug 1978. Contract EY-76-C-03-0893-031. 102p. AT.

Analysis of the Advanced Fuels Critical Experiments in ZPR-9 has been completed. The experimental program, jointly planned by ANL and GE, contained three critical configurations identified as the Carbide Benchmark, Carbide Zone, and Advanced Oxide Zone Assemblies. These experiments provided data to examine the relative breeding, core performance, and safety parameters of advanced fuel materials. Analysis using normal LMFBR design techniques shows that carbide and oxide fueled cores can be evaluated with equivalent accuracy. Generally good agreement between calculated and measured parameters is observed.

- 1288 (GEFR--00394) OPTIMIZED SPENT FUEL STORAGE AND SHIPPING SYSTEMS. Davis, C.R. (General Electric Co., Sunnyvale, CA (USA)). Aug 1978. Contract EY-76-C-03-0893-030. 138p. AT.

This report describes the development of optimum, integrated, Liquid-Metal Cooled Reactor (LMCR) spent fuel storage and shipping systems. The requirements are described for facility interfaces and major component design features in order to have efficient, effective, minimum cost spent fuel storage and shipping systems. Due consideration is given to various refueling intervals, reload batch sizes, and recycle times, since the out-of-core fuel inventory has a pronounced influence on fuel cycle costs.

- 1289 (GEFR--00395) STUDY OF TRANSIENT SODIUM BOILING IN LMFBR FUEL BUNDLES. Shih, T.A. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Apr 1979. Contract EY-76-C-03-0893-032. 78p. AT.

An assessment of the dynamics of sodium boiling in LMFBR fuel bundles undergoing transient undercooling events has been performed and is reported. Since there is no suitable two-dimensional analytical tool available for such evaluation, the analysis was conducted using special overlays of existing

computer codes and approximation methods. An attempt was also made to develop a simple multi-channel model which is capable of predicting axial and radial boiling progression and the time of inlet flow reversal. Analytical predictions obtained with this model were compared with available experimental results and were judged satisfactory for use in near-term safety test design engineering.

- 1290 (GEFR--00396) DEVELOPMENT AND PRELIMINARY VERIFICATION OF CORTAC-3D. Fox, J.N.; Wei, J.P. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-031. 283p. AT.

The report documents the development and preliminary verification effort associated with the CORTAC-3D computer code. CORTAC-3D is a three-dimensional version of the two-dimensional CORTAC computer code which was made available for general use in mid-1976. The CORTAC family of computer codes calculates the short-term steady-state and transient response of an LMFBR core restraint system. This requires the simultaneous analysis of the coupled effects of reactor core thermal-hydraulics, component structural deflections, and reactor kinetics. In spite of the demonstrated general usefulness of the two-dimensional version of CORTAC, several significant 3-D effects were identified in FY-77 which provided the incentive to develop a 3-D version of the code. These effects included: (1) thermal-hydraulic modeling shortcomings, numerical difficulties, verification concerns, and exclusion of important subchannel thermal-hydraulic details; (2) "averaged" rather than discrete thermal bowed shapes and load pad forces; (3) regional rather than row-wise Doppler, axial expansion and material densities reactivity feedback calculations. Since 3-D reactivity feedbacks are a simple extension of those employed in CORTAC (the point kinetic model will be identical) and 3-D structural modeling is available with NUBOW-3D, the major 3-D development item is clearly the thermal-hydraulic model. Hence, the majority of the FY-78 effort was directed toward the development and verification of the thermal-hydraulic portion of CORTAC-3D.

- 1291 (GEFR--00397) DEVELOPMENT OF BACKUP REFUELING METHODS. Fox, J.N. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-031. 16p. AT.

The influence of long-term irradiation-induced material property effects, coupled with the complexity of analyzing a three-dimensional array of parallel core assemblies with small gaps at the load pad planes, make it extremely difficult to predict accurately the load pad normal forces and ultimately the force required to withdraw or insert an LMFBR core assembly at refueling. The accuracy of such load predictions is also subject to the uncertainties in the operational environment (flux and temperature distribution), creep and swelling correlations, and the uncertainties of pad-to-pad and nozzle-to-receptacle friction coefficients. Hence, it is not unreasonable to expect that actual withdrawal loads will exceed core assembly withdrawal force design limits. Therefore it is important to develop backup refueling method that can reduce core assembly withdrawal or insertion loads to acceptable levels during the refueling process. These methods provide a secondary or backup means for completing assembly refueling without damage to neighboring assemblies or the core assembly receptacles. The backup refueling methods

activity has included proof-of-principle testing, development of backup refueling devices, and the development of backup refueling procedures. Each of these efforts is described in the report.

- 1292 (GEFR--00398) CARBON MIGRATION IN TRANSITION JOINT WELDS. Mitchell, M.D.; Offer, H.P.; Ring, P.J. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-30. 97p. AT.

Diffusion and mixing are known to occur at the fusion boundary of transition weld joints, particularly in certain austenitic/ferritic material combinations where the weld and base metal have different carbon diffusivities and carbon activities. Heat from the welding process and post weld heat treatment accelerates carbon diffusion, generating adjacent carburized and decarburized bands. A program was conducted to quantify the effects of weld metal dilution, time, and temperature on the carbon migration at the fusion zone of the 2 1/4 Cr-1Mo steel/Inconel 82 weld interface in a CRBRP prototypic transition joint. The electron microprobe was used to measure chemical gradients, particularly the carbon gradient, across the weld interface. The major emphasis of the study was placed on establishing a time-temperature relationship which could predict the extent of carbon migration under service conditions from short time, high temperature tests. Based on the similarity of carbon profiles obtained from samples treated at 510C (950F) for 1000 hours and 677C (1250F) for 10 hours, a Larson-Miller type equivalency was employed to establish the time-temperature relationship.

- 1293 (GEFR--00399) POSTIRRADIATION EXAMINATION RESULTS FROM EXPERIMENT F11A (EBR-II SUBASSEMBLIES X141 AND X141A). Rosa, J.M. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-031. 69p. AT.

The F11A experiment has achieved a peak burnup of 7 atom % in subassemblies X141, X141A, and X141B. Two fuel pins, operating at peak cladding temperatures in excess of 1350°F, experienced cladding breach. The first of these was the first breach of an unencapsulated fuel element in a GE experiment. Post-irradiation examination data is reported on five fuel pins irradiated in subassemblies X141 and X141A, including the initial breach. The cladding breach is a thermal creep rupture probably caused by local intergranular fuel-cladding chemical interaction (FCCI). Alteration of cladding morphology (precipitation of intergranular sigma phase) is evident near the breach. Metallographic examination of companion pins which operated at lower cladding temperatures show sound cladding with insignificant FCCI.

- 1294 (GEFR--00400) INTERIM REPORT: LEAK DETECTION TEST ACTIVITIES ON THE GENERAL ELECTRIC STEAM GENERATOR TEST RIG (SGTR). Stehling, S.G.; Magee, P.M. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-030. 71p. AT.

Leak detection equipment and their associated control and instrumentation systems have been developed under Base Development Program SG027, Subtask X1, to detect the reaction products which evolve from the reaction of sodium with small quantities of water in LMFBR steam generators. The prototype leak detector characterization tests which have

been initiated in the General Electric Steam Generator Test Rig (SGTR), in conjunction with the Few Tube Test (FTT) Program, are the first in a series of tests designed to assure the adequacy of the system design for commercial LMFBR plant utilization. The SGTR tests are the first time the prototype Clinch River Breeder Reactor (CRBR) design leak detectors are tested, and the primary emphasis of the tests is on system operation and procedure checkout, with secondary emphasis placed on obtaining signal resolution and transit time data.

- 1295 (GEFR--00402(Vol.1)) PRE-IRRADIATION CONCEPT DESCRIPTION AND COST ASSESSMENT. Pflasterer, G.R.; Deane, N.A. (comps.). (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-031. 35p. AT.

The objective of this study is to develop a Pre-Irradiation Facility (PIF) design concept, and a preliminary capital and operating cost assessment for this facility, for the purpose of estimating the effect of pre-irradiation on the LMFBR energy cost.

- 1296 (GEFR--00402(Vol.2)) PRE-IRRADIATION CONCEPT DESCRIPTION AND COST ASSESSMENT. Pflasterer, G.R.; Deane, N.A. (comps.). (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-031. 276p. AT.

A concept evaluation and cost assessment for a water-cooled reactor facility for the pre-irradiation of LMFBR fuel assemblies is described. This study is part of the LMFBR advanced core design program aimed at PLBR development. Volume II contains chapters on the reactor description, plant description, cost assessment, impact of pre-irradiation on LMFBR capital cost, impact of pre-irradiation on LMFBR total energy costs, and demonstration of pre-irradiation concept progress.

- 1297 (GEFR--00402(Vol.3)) PRE-IRRADIATION CONCEPT DESCRIPTION AND COST ASSESSMENT. Pflasterer, G.R.; Deane, N.A. (comps.). (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-031. 455p. AT.

The various appendices cited in Volumes I and II are included in this volume.

- 1298 (GEFR--00403) C-1 IRRADIATION CREEP IN BENDING EXPERIMENT: SUMMARY DESCRIPTION AND RESULTS OF THE FIRST TWO INTERIM EXAMINATIONS. McSherry, A.J.; Marshall, J.; Patel, M.; Ring, P.J.; Konze, G.K.; Appleby, W.K. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-031. 77p. AT.

The design constraints and a description of the final design of an irradiation experiment to measure irradiation creep in bending of 20% cold worked AISI 316 stainless steel is presented along with the results of the first two interim examinations. Analysis of this low fluence data indicates that irradiation creep in bending has a larger transient creep component than observed in biaxially stressed specimens, but the secondary (steady state) creep rates appear similar and will be quantified by subsequent higher fluence measurements.

- 1299 (GEFR--00404) CLADDING COMPONENT CHEMICAL TRANSPORT IN MIXED-OXIDE FAST REACTOR FUEL PINS: AN ADDITIONAL (HIGH BURNUP) MECHANISM FOR FUEL CLADDING CHEMICAL

INTERACTION. Adamson, M.G.; Aitken, E.A. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 19 Sep 1978. Contract EY-76-C-03-0893-031. 52p. AT.

Transport of the principal metallic components of Fe-Cr-Ni austenitic cladding alloys into the fuel phase of irradiated oxide fuel pin has been reported periodically since the inception of fuel pin testing in the Fast Reactor Development Program, however, thus far, an adequate explanation for this phenomenon has been lacking. In this document we review available in-pile evidence for cladding component transport in oxide pins, delineate the major variables that appear to be responsible for its occurrence, outline the development of a working physicochemical model for cladding component chemical transport (CCCT) at moderate to high burnups, and describe how such a model may be applied in the calculation of cladding wastage due to component transport. The basis of the CCCT mechanism for FCCI is a liquid Cs,Te fission product medium in the fuel-cladding gap in which Fe, Cr, Ni, Mn, etc., can dissolve and undergo transport up the steep (radial) temperature gradient. Special requirements on the quantities of Te (and Cs) fission products present in the gap explain why CCCT is generally observed only at moderate to high burnups, and characteristic morphological features of CCCT are consistent with a dissolution-type mechanism. The thermodynamic basis of the CCCT model is also treated in some detail, and it is shown, for example, how oxygen activity influences the equilibria between tellurium and dissolved cladding components in the Cs,Te fission product melt. A preliminary quantitative analysis of recent in-pile data is performed and the results used to derive coefficients for some of the key parameters in the rate model. Finally, the model is applied to a long, commercial LMFBR-type fuel pin to illustrate the expected important contribution of CCCT to FCCI-induced cladding inner surface wastage at high burnup.

- 1300 (GEFR--00405(Vol.1)) IMPROVEMENTS TO THE MAINTAINABILITY AND INSPECTION OF LIQUID METAL SYSTEMS. VOLUME 1. OVERALL STUDY RESULTS. Sciacca, F.W.; Scott, W.B. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-030. 567p. AT.

The Systems Maintainability and Inspection Improvements Study was a subtask of the Liquid Metal Systems Development Studies (DOE Contract EY-76-C-03-0893, 189a SG066). This subtask had the objective of identifying and evaluating improvements to the maintenance and inspection of the reactor, heat transport, and steam generation systems of loop-type liquid metal cooled reactor (LMCR) plants. More specifically, the aim of the effort was to identify concepts and ideas to accomplish the following: reduce the time for maintenance and inspection; improve the quality of maintenance and inspection; reduce personnel exposure; increase plant availability; improve plant constructability; and reduce total energy costs.

- 1301 (GEFR--00405(Vol.2)) IMPROVEMENTS TO THE MAINTAINABILITY AND INSPECTION OF LIQUID METAL SYSTEMS. VOLUME 2. PLANT LAYOUT IMPROVEMENTS. Golan, S. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.; Bechtel National, Inc., San Francisco, CA (USA)). Sep 1978. Contract EY-76-C-03-0893-030. 137p. AT.

A study has been performed for General Electric to determine the technical feasibility

and cost of arranging loop-type LMFBR primary heat transport cells to incorporate features that facilitate in-service inspection and maintainability. Specific areas requested to be examined were: primary heat transport system (HTS) piping layouts; HTS cell configurations; a shield floor; routing of auxiliary services in the HTS cell; maintenance equipment space requirements; effect on plant cost and construction schedule of alternative layout; and effect on plant constructability of alternative layout. A study was also performed to investigate the feasibility of primary HTS modularization as a means to improve plant constructability and maintainability.

- 1302 (GEFR--00406) CORE PERFORMANCE EVALUATION OF ADVANCED CLAD/DUCT ALLOYS. Branlund, B.J. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 1 Sep 1978. Contract EY-76-C-03-0893-031. 139p. AT.

The objective of the LMFBR core materials performance evaluation was to provide rankings to the Clad/Duct Working Group to assist their screening of the candidate alloys (D9, D11, M813, PE16, D21, D25, D66, INC706, D68, HT9, and D57). The working group was attempting to reduce the number of candidate alloys to six. The six most attractive alloys are D9, PE16, D66, D68, HT9 and D57 based on the overall core cost rankings provided by COROPT optimization studies. For 40% of the cases considered, five of these six alloys were no more than 10% more expensive than the top ranked alloy; all other alloys fell within the 10% range in less than 15% of the cases.

- 1303 (GEFR--00407) IRRADIATION AND POST-IRRADIATION EXAMINATION DATA FROM SELECTED F9E FUEL PINS. Plumlee, D.E. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-031. 90p. AT.

The design, operating parameters, and post-irradiation examination results are presented from five mixed-oxide fuel pins (F9E02, F9E04, F9E23, F9E41, and F9E42) irradiated in the General Electric F9E experiment. The F9E experiment was designed as a diagnostic test to assess subassembly performance for fuel pins spaced using several candidate grid spacer designs. The companion F9 tests (F9A, B, C, and D) containing pins spaced by wire wrapping provided the reference base for judging grid spaced fuel pin performance. Subassembly and grid spacer performance have been previously reported.

- 1304 (GEFR--00409) EVALUATION OF PROTOTYPICAL CRBRP STEAM GENERATOR TUBE/TUBESHEET WELDS. Offer, H.P.; Ring, P.J. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-030. 43p. AT.

The CRBRP tube/tubesheet welds are being evaluated for quality and uniformity of properties. This evaluation is proceeding on a continuing basis throughout the fabrication cycle of the CRBRP steam generators. The weld evaluation includes bend tests to identify susceptibility to brittleness, hardness tests to determine the effectiveness of post weld heat treatments in reducing the weld strength, stress corrosion cracking tests to evaluate caustic stress corrosion susceptibility under faulted conditions, flexural fatigue tests (to characterize the effects of the weld geometry, metallurgical structure and allowable porosity), and stress rupture and burst tests to evaluate the weld mechanical efficiency

under biaxial stress, relative to the tubing strength. Metallography and nondestructive testing techniques are also used to help in interpretation of the mechanical test results and to aid in verifying the weld quality and uniformity.

- 1305 (GEFR--00411) COMPENDIUM OF CORE DEBRIS RETENTION TECHNOLOGY. Gluekler, E.L. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 29 Sep 1978. Contract EY-76-C-03-0893-032. 108p. AT.

A summary is presented of core debris retention technology pertinent to demonstration of successful control of accident progression for postulated core melting events in liquid metal fast breeder reactors. Data and models in key areas of in-vessel and ex-vessel debris retention are reviewed, including decay heating rates, heat source distribution, potential for secondary criticality, debris fragmentation and settling characteristics, coolability of particle beds, molten layers and boiling layers, primary system integrity, ex-vessel material interactions, debris penetration into concrete, and containment responses. Some concepts applied to early liquid metal fast breeder reactors for consequence mitigation are described, and recommendations are made for future work needed to support demonstration of successful debris retention.

- 1306 (GEFR--00413) USSR STEAM GENERATOR MATERIALS EXAMINATION OF TEST SPECIMENS. Hampton, L.V.; Roy, P. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jan 1979. Contract EY-76-C-03-0893-031. 54p. AT.

Metallography, radiography, chemical analysis and electron-microprobe analysis were used to characterize samples of USSR steam generator materials and sodium loop specimens. The samples were analyzed with regard to steam side corrosion and deposition, decarburization, structure, composition and weld integrity. The lack of suitable archive materials and precise operating conditions limits the number of firm conclusions which can be drawn as a result of the examination. An attached Appendix lists all the available initial data and the type of specimens examined.

- 1307 (GEFR--00414) BASELINE INSPECTION OF THREE TRANSITION JOINT LIFE TEST ARTICLES. Day, R.A. (General Electric Co., Sunnyvale, CA (USA)). Dec 1978. Contract EY-76-C-03-0893-030. 48p. AT.

Ultrasonic inspection was accomplished on the Transition Joint Life Test (TJLT) articles using a semiautomatic computer controlled pipe scanner. Tests were capable of reliable detection of ten and five percent notch in the Alloy 800 and 316 stainless steel parent materials and at least a three percent notch in the 2-1/4Cr--1Mo parent metal. The detection of notches in the weld metal has not been established, but ten percent notches were detectable above weld metal noise in the 316 parent metal by insomifying the 16-8-2 weld metal from the Alloy 800 parent metal. The TJLT joints will be reinspected at intervals at the Energy Technology Engineering Center (ETEC) in Los Angeles to detect creep fatigue cracks in the 2-1/4Cr--1Mo HAZ. The same equipment will be used with some improvements to do the in-service inspections.

- 1308 (GEFR--00415) IN-VESSEL CORE DEBRIS COOLABILITY FOR LARGE BREEDER REACTORS. Yang, H.K.; Gluekler, E.L. (General Electric Co.,

Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 31 Oct 1978. Contract EY-76-C-03-0893-032. 64p. AT.

Work is in progress to evaluate the ability of inherent structural barriers and natural processes to mitigate the consequences of extremely low probability events which could involve meltdown of a portion of the reactor core. In this report, an assessment of the coolability of core debris within the primary system boundary of large breeder reactors is presented. Since the state and configuration of fuel after neutronic shutdown by relocation from the core region is uncertain, the coolability limits were determined for debris in the form of small fragmented particles, as well as solid or molten layers. To assess the coolability of fragmented particles, it was necessary to extend existing analytical models for particle bed dryout to include the effects of heat removal from the bottom of the particle layer, and sodium subcooling. At 1000 seconds after neutronic shutdown, the maximum volume fraction of core fuel that can be cooled as fragmented particles was predicted to be on the order of 5%. A flat retention plate located underneath the core support structure would improve the particle coolability limit to volume fractions between 50 and 100% depending on the design configuration. The analysis of solid and molten layers indicated that dryout and remelting of debris particle layers do not necessarily lead to debris penetration through the primary system boundary, if the following conditions are met: (a) heat must be removed from the lower surface of the structure retaining the debris by sodium natural convection, or (b) the structure must be able to withstand surface temperatures close to its melting point. Otherwise, the coolability limit for solid and liquid layers was found to be on the order of 0.25 to 0.7% of the core volume, depending on the final debris configuration and the boundary conditions assumed.

- 1309 (GEFR--00417) PACT 2: PROBABILISTIC ANALYSIS OF COOLANT AND CLADDING TEMPERATURES. Chen, K.H.; Hoover, D.G. (General Electric Co., St. Petersburg, FL (USA). Pinellas Peninsula Plant). Jun 1978. Contract EY-76-C-03-0893-031. 239p. AT.

The PACT 2 code computes the probability distributions of the coolant and cladding temperatures throughout a single LMFBR fuel assembly during steady state operation at start of life. The PACT 2 code is an improved version of PACT. It uses a mathematical model that considers both thermal-hydraulics in coolant subchannels and Monte Carlo sampling of the thermal-hydraulic calculation. The coolant temperature in each nodalized coolant subchannel is defined as a function of enthalpy, flowrate, crossflow between adjacent subchannels, heat flux, and position. The mathematical model neglects the redistribution of subchannel coolant flow; therefore, it is limited to fuel pin assemblies in which there is no flow blockage. The equations of the mathematical model for the thermal-hydraulic calculation are solved by using a forward marching finite difference scheme.

- 1310 (GEFR--00419) PRELIMINARY MODELS FOR EVALUATING RADIOLOGICAL SOURCE TERMS. Huang, T.C.; Dayan, A.; McDonald, J.W.; Teresi, J.D. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 17 Nov 1978. Contract EY-76-C-03-0893-032. 48p. AT.

The purpose of the safety work described herein is to develop the technology for evaluating mechanisms that attenuate radiological consequences. This report presents model

developments from October 1977 through September 1978 (FY 1978) and indicates experimental needs regarding model development and verification. The work covers modeling of extremely low-probability accident scenarios resulting in fuel vaporization and discharge of sodium and core debris to the containment building, the generation of melt-through source terms by sparging and vaporization, and the attenuation of energetics and melt-through source terms.

- 1311 (GEFR--00422) TRANSITION JOINT WELDING: LIFE TEST ARTICLE FABRICATION. Hartle, R.T.; Busboom, H.J.; Ring, P.J. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Dec 1978. Contract EY-76-C-03-0893-030. 90p. AT.

This work is in support of the Clinch River Breeder Reactor (CRBR), a demonstration plant funded jointly by the Department of Energy and the Project Management Corporation. The objective of this task was to improve the conventional transition joint design and develop welding fabrication techniques for these improved joints. The joints for CRBR form the sodium side transitions between ferritic and austenitic stainless steel piping within the Intermediate Heat Transfer System (IHTS). These joints vary in size from 1 inch, Schedule 80 up to 26.0 inches, and are required to perform successfully throughout a design life of 30 years (approx. 263,000 hours).

- 1312 (GEFR--00423) CORE RESTRAINT SYSTEM DESIGN GUIDELINES. Fox, J.N. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jan 1979. Contract EY-76-C-03-0893-031. 64p. AT.

Guidelines contained herein were developed by examining the fundamental core restraint functional criteria which require that the core restraint system: (1) establish and control the radial and azimuthal position of the reactor core assemblies such that reactivity variations from radial and azimuthal motion will not limit safe reactor operation under all credible conditions; (2) maintain critical alignments between in-core and above-core components; and (3) permit refueling within specified load limits. While these functional criteria are important, they do not provide sufficient guidance to the individual designer (contractor) to develop detailed design criteria that are generally acceptable to the LMFBR community. This is due primarily to the complex behavior of the core restraint system, its many interfacing systems, and the individual interpretation of safe reactor operation. Hence a set of comprehensive core restraint system design guidelines are necessary that not only address the functional criteria, but reflect design and safety philosophies that are consistent with current core restraint engineering technology. These design guidelines, once endorsed by the CM and CP working group will allow each core restraint system designer to develop consistent and well-based detailed design criteria.

- 1313 (GEFR--00424) PHYSICOCHEMICAL MODEL FOR PREDICTING SODIUM REACTION SWELLING IN BREACHED LMFBR FUEL AND BREEDER ELEMENTS. Caputi, R.W.; Adamson, M.G.; Evans, S.K. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Mar 1979. Contract EY-76-C-03-0893-031. 48p. AT.

The use of sodium as a coolant for the Liquid Metal Fast Breeder Reactor (LMFBR) necessitates study of the properties of the coolant and chemical interactions it may

undergo with the reactor fuel. If a defect or breach develops in the cladding of an irradiated fuel pin, sodium is likely to enter the pin and react chemically with the fuel. This reaction results in formation of a sodium-fuel reaction product which, since it has a lower density than the fuel, can cause the fuel pin to swell. An investigation is presented for the reaction of sodium with the mixed-oxide fuel following a breach of the cladding. Predictive correlations are presented for swelling with time in breach fuel rods that will be helpful to those concerned with performance of breached driver fuel pins and blanket rods in an LMFBR and plant-related consequences.

1314 (GEFR--00426-1) HYDRAULIC TESTING PROGRAM ON VENTED ASSEMBLY CORE DESIGN, FY-78. VOLUME 1: VENT HYDRAULICS TESTING. Edelfelt, I.H. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jan 1979. Contract EY-76-C-03-0893-031. 22p. AT.

During FY-77 a combined theoretical and experimental program was conducted for the purpose of determining the key parameters for the vent hole design in an actual LMFBR. The theoretical work was based on the pipe break flow analysis as modeled by Quan. This analysis predicted two types of flow regimes; free jet and confined flows, with the particular classification depending upon the relative area of the vent hole to that of the channel gap. The vent area is the cross-sectional area of the vent hole, while the gap area is the area resulting by multiplying the vent perimeter by the duct-to-duct (interstitial) spacing. In the case of a small vent-hole-to-gap area ratio such that the vent flow is limited by the hole area, a free jet model was used. On the other hand, if the channel gap area is smaller than that of the vent area, a confined flow model was used. The FY-78 effort which is being reported here included two vent flow pressure drop tests, both performed with a constant 1.5-inch diameter vent size. The first consisted of rotating the test channel in the test rig by 60 degrees so that the vent holes would be opposite one another. This configuration was tested over the same range of variables as was used in the FY-77 tests. The second set of tests employed a reverse, or inflow, configuration wherein the channel was in its standard orientation with the vents not opposite one another but with flow out of the interstitial space into the tube bundle.

1315 (GEFR--00426-2) HYDRAULIC TESTING PROGRAM ON VENTED ASSEMBLY CORE DESIGN, FY-78. VOLUME 2: PIN VIBRATION TESTING. Edelfelt, I.H. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jan 1979. Contract EY-76-C-03-0893-031. 224p. AT.

One problem which could conceivably exist in any reactor assembly and which, therefore, must be investigated is flow induced vibration of the fuel pins. It is to the evaluation of this possible source of trouble that the work reported here was addressed. During FY-77, a combined theoretical and experimental program was conducted for the purpose of determining the flow and pressure drop characteristics for the vent hole design in an actual LMFBR. During the 1977 program, a flow loop was constructed incorporating a test model. The fuel pin vibration study reported herein makes use of this test model and flow loop. The model consists of a single channel containing a tube bundle. Surrounding this channel was another hexagonal channel which duplicated the adjacent

walls of the six channels which would normally surround any interior channel in a complete fuel assembly. The spacing between the two channels thus corresponded to the interstitial gap. A biaxial accelerometer was installed in each of eight tubes, or pins as they are referred to in an actual reactor unit; four adjacent to vents, two adjacent to nonvented walls and two near the center. In all cases, the accelerometers were located in the plane of the three vent holes. Data were recorded from the accelerometers for the no flow case, for reference purposes, and with inlet flows of 240 GPM, 320 GPM and 400 GPM with vent flow from 0 to 30% of the inlet flow. The results were processed by a Fourier analyzer to yield plots of g loading versus frequency. The resulting plots are given in this report along with a discussion of the results.

1316 (GEFR--00431) INFLUENCE OF DECARBURIZATION ON THE 510°C (950°F) TENSILE AND CREEP PROPERTIES OF THIN SECTION 2-1/4CR--1MO STEEL WELDMENTS. Mandurrango, G.L.; Pizzo, P.P. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Feb 1979. Contract EY-76-C-03-0893-031. 42p. AT.

The short-term tensile and creep properties of severely decarburized 2-1/4Cr--1Mo steel autogenous weldments at 510°C were determined by the properties of the base material. Limited deformation occurs within the autogenous weld region. Base metal properties will provide conservative design rules for CRBR. Microstructural damage, such as grain boundary void growth and/or triple point cracking, was not observed in either the Weld, HAZ, or Base Metal regions for conditions scoped in this study. Abnormal grain growth occurred within weld regions for the more extreme decarburization conditions. Abnormal grain growth was similarly observed in normalize and tempered (NT) specimens. Isothermal anneal and post weld heat treatment (IA + PWHT) specimens did not exhibit abnormal grain growth. Based on the limited kinetics data available, initiation of abnormal grain growth in decarburized regions will not occur in a design life of 30 years for temperature lower than 520°C (968°F).

1317 (GEFR--00432) MATERIALS CHARACTERIZATION: STUDY OF DECARBURIZATION INFLUENCE ON MECHANICAL PROPERTIES OF CRBR STEAM GENERATOR TUBING. Pizzo, P.P. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Feb 1979. Contract EY-76-C-03-0893-031. 45p. AT.

Tensile characterization tests at 510°C confirm that the thin section coupon specimens selected for the decarburization study exhibit analogous data trends to CRBR steam generator tubing. A real secondary size/geometry factor must be applied in interpreting steam generator tube properties from coupon data. Tensile ductility and ultimate tensile strength values require adjustment. Baseline creep data for tubular specimens selected for decarburization study correlate well with the Nuclear Systems Materials Handbook (NSMH) time-to-rupture and minimum creep rate correlations for annealed 2-1/4 Cr--1Mo steel. Thermal aging 1000 hours at 510°C reduces the tensile strength of 2-1/4 Cr--1Mo steel by approximately 17%. The effect is intrinsic. Further data are required to assess implications with respect to design. Scoping tests indicate that cold work enhances creep resistance of 2-1/4 Cr--1Mo steel while maintaining creep ductility. Fabrication induced cold work has significant impact on creep behavior, and a pretest stress relieving treatment must be incorporated in the

decarburation study. Use of UTS as a correlation parameter for creep behavior is not supported by data. Although 510°C thermal aging reduces the UTS of 2-1/4 Cr-1Mo steel 17%, the creep properties before and after 510°C thermal aging are equivalent. Mechanical properties variations noted could not be correlated with carbide size and distribution. Thin film transmission electron microscopy techniques are required to provide structure/property correlation. Creep cavitation was observed at 510°C and at a stress of 130 MPa. Rupture time and minimum creep rate values were 11,253 hours and $1.6 \times 10^{-3}\%$ per hour. Further data are required to determine the nature of the cavities and to assess the impact of cavity formation on the resultant creep behavior.

- 1318 (GEFR--00436) FULMIX-II: A COMPUTER PROGRAM FOR STEADY-STATE THERMAL-HYDRAULIC ANALYSIS OF WIRE WRAPPED ROD BUNDLES. Wei, J.P. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 1 Mar 1979. Contract EY-76-C-03-0893-031. 250p. AT.

This report provides a description of the FULMIX-II computer program, detailed instructions for its use, and a brief discussion of the features of the program. FULMIX-II is a modified version of FULMIX [I]. FULMIX-II is designed specially to determine single-phase, steady-state coolant flow and temperature distributions in a wire wrapped assembly with reference wire wrap design (straight starts) and other wire wrap configurations (staggered starts, wireless rods, locked wrap, etc.). The code can be used for hexagonally arranged assemblies with 7 to 271 fuel rods. The code performs the steady-state calculation of average temperatures for each fuel rod and coolant flow subchannel in the assembly at an arbitrary number of axial planes. The thermal-hydraulic analyses presented in this report are based on application of the subchannel concept in association with the use of bulk parameters for coolant velocity and coolant temperature within a subchannel. The interactions between subchannels are due to turbulent interchange, pressure-induced crossflow, directed sweeping crossflow induced by the helical wire wrap, and transverse thermal conduction. Flow field in the pin bundle is solved first using an iterative method in a finite difference scheme. Once the flow field is determined, the energy equation is decoupled, and the temperature distribution is calculated accordingly.

- 1319 (GEFR--00439) BUNDLE TO DUCT INTERACTION COMPRESSION TESTS USING LARGE DIAMETER PINS. Kaplan, S. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 19 Mar 1979. Contract EY-76-C-03-0893-031. 88p. AT.

Bundle compression testing was recently performed on large diameter pin bundles to expand the bundle compression test data base previously obtained using small diameter (0.230 inch) pins. The large diameter pin bundles used the carbide fuel pin size (0.370-inch diameter) and CRBR radial blanket pin size (0.506-inch diameter). The recent testing included pin ovaling measurements, data not obtained during the smaller diameter pin tests. This report summarizes the compression testing performed on pin bundles using the large diameter pins.

- 1320 (GEFR--00441) FLOW-INDUCED VIBRATION AND HYDRAULIC TEST OF THE FTR 00-00-00/WIRELESS ADVANCED WIRE-WRAP FUEL BUNDLE ASSEMBLY. Lee, K.L. (General Electric Co., Sunnyvale, CA

(USA). Advanced Reactor Systems Dept.). Mar 1979. Contract EY-76-C-03-0893-031. 133p. AT.

A series of vibration and hydraulic tests were performed on a mock-up Fast Test Reactor (FTR) fuel assembly for the Fast Flux Test Facility (FFTF) to assess the vibrational and hydraulic characteristics under simulated flow conditions. The 217-pin advance wire wrap (00-00-00/Wireless) fuel assembly was tested under simulated reactor flow conditions using demineralized water to simulate the primary sodium coolant. An adjustable (across-flats) duct was used to vary the bundle-duct assembly clearance (porosity). The tests were required because of the difficulty in performing accurate analytical predictions of the fuel assembly performance due to the turbulent flow conditions and highly complex structural boundary conditions. The test results confirmed that the fuel pins met the FTR design criteria for vibrations. The fuel pin displacements were less than 2 mils (rms). The vibration behavior was similar to wire wrapped assemblies. The fuel pin response spectra were characterized by distinct, highly damped resonances at low flow rates. As the flow rate increased, the fluid-forced response dominated until at high flow rates the resonant response was almost completely masked by broad band fluid forces. Vibrations were relatively independent on bundle porosity but flow friction losses were definitely dependent on bundle porosity.

- 1321 (GEFR--00442) TWOD-2: TWO-DIMENSIONAL FINITE ELEMENT FUEL PIN ANALYSIS CODE. Patel, M.R. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 23 Mar 1979. Contract EY-76-C-03-0893-031. 79p. AT.

TWOD-2 is a two-dimensional finite element code developed specifically for thermostructural analysis of Fast Breeder Reactor (FBR) fuel pins subjected to normal operation and design basis transients. It enables evaluation of effects of geometrical and environmental asymmetries and thus complements the LIFE system of one-dimensional codes being developed as the reference methods for evaluating the performance of FBR fuel pins. The code permits analysis of heat transfer, and elastic-plastic-creep-swelling and thermal deformations in isotropic solids deforming in plane stress, plane strain and axisymmetric modes. Fuel-cracking, crack-healing, fuel-cladding gap closure and friction at the fuel-cladding interface are modeled by permitting orthotropic elasticity and conductivity in the elements. The current version models (U,Pu) O₂ - 26CW316SS behavior.

- 1322 (GEFR--00444) TRANSWRAP-II ANALYSES OF THE STANFORD RESEARCH INSTITUTE IHTS/RELIEF SYSTEM SIMULATION TESTS. Wolfe, C.F. Jr. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Apr 1979. Contract EY-76-C-03-0893-030. 254p. AT.

Tests were performed by the Stanford Research Institute (SRI), Menlo Park, California, in which the time histories of acoustical pressure pulses were measured at various locations in a water loop that simulated a LMFBR intermediate heat transport and pressure relief system. This report focuses on the computer simulation of these experiments, and the comparison of the analytical results to corresponding measured values. The report shows that TRANSWRAP has the capability to predict adequately the pressure response of the SRI scale model of an LMFBR IHTS. Fluid velocities in the relief line are shown to be

conservatively predicted.

- 1323 (GEFR--00445) ANALYSIS OF THE ZPPR-8 CRITICAL EXPERIMENTS. Lu, Y.S.; Crick, S.C. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Mar 1979. Contract EY-76-C-03-0893-031. 83p. AT.

The purpose of this report is to present the analysis of a series of experiments performed in the Zero Power Plutonium Reactor (ZPPR), identified as ZPPR Assembly 8. This assembly simulates a clean condition of the demonstration-reactor class LMFBR core with a heterogeneous (radial parfait) reference configuration which uses thorium in the internal blanket regions. The experimental program in ZPPR-8 was specified to provide a benchmark study of the neutronic behavior of a heterogeneous LMFBR core with an emphasis on the impact of a thorium blanket. A thorium blanket was loaded at different locations for various phases of ZPPR Assembly 8. This report presents details on the experimental program, criticality, core midplane reaction rate distributions, sodium void and subassembly worth measurements, control worth measurements and loading error simulations, and finally the conclusions.

- 1324 (GEFR--00446) EFFECTS OF THERMAL AGING ON THE MECHANICAL PROPERTIES OF A 2-1/4 CR-1MO STEEL/INCONEL 82 TRANSITION JOINT FUSION ZONE. Mitchell, M.D.; Offer, H.P. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Apr 1979. Contract EY-76-C-03-0893-030. 90p. AT.

Transition joint weldments between austenitic and ferritic materials are required in the CRBRP Project piping systems. Annealed 2-1/4Cr-1Mo steel welded with ERNiCr-3 (Inconel 82) to Incoloy 800H using the hot wire automatic gas tungsten arc process will be used for a number of these joints. This effort provides a scoping study of the mechanical properties at the 2-1/4Cr-1Mo steel/Inconel 82 fusion zone of these joints to provide actual material properties for the TJLT inelastic analysis and to provide a better understanding of the probably degradation of properties due to thermal aging and carbon migration as a function of the time at temperature. Elevated temperature, tensile, stress rupture, strain controlled fatigue, and fatigue crack growth tests were performed. Thermal aging at 677°C (1250°F) for ten hours (equivalent to approximately 1000 hours at 510°C (950°C), the nominal service temperature), was found to reduce the tensile and stress rupture properties of the weldment at the fusion zone. The fatigue crack growth rate and continuous fatigue life at the fusion zone were not affected by the thermal aging; however, this zone was shown to crack more readily than either the annealed 2-1/4Cr-1Mo steel or the Inconel 82 weld deposit.

- 1325 (GEFR--00449) FOULING AND CHEMICAL CLEANING STUDIES: CRBRP EVAPORATORS. PART II. Hampton, L.V. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). May 1979. Contract EY-76-C-03-0893-030. 39p. AT.

Tests were conducted to study the chemical descaling process of removing fouling deposits from the waterside surfaces of sodium-heated 2-1/4Cr-1Mo steel evaporator tubes. Test solutions were ammoniated EDTA (ethylenediamine tetraacetic acid), ammoniated citric acid and ammoniated EDTA + citric acid. The fouling product layers were characterized by both energy dispersive x-ray and electron microprobe

techniques. Tests were conducted at approximately 80°C (176°F) and at 25.2 cm³/sec (0.4 GPM) and 252 cm³/sec (4.0 GPM). The base metal corrosion rates during cleaning were determined. Ammoniated EDTA + citric acid solutions yielded the most acceptable results.

- 1326 (GEFR--00463) STATE-OF-THE-ART WATER CHEMISTRY OF STEAM GENERATORS WITH EMPHASIS ON CRBRP SPECIFICATIONS. Simpson, J.L.; Dutina, D.; Robles, M.N.; Spalaris, C.N. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jun 1979. Contract EY-76-C-03-0893-030. 62p. AT.

This report is an Addendum to Reference 1, presenting an update of the CRBRP water chemistry specifications, as well as the supporting data and discussion for the changes made. Experimental data are presented from three tests conducted by General Electric. A resolution was obtained to relate water chemistry and the occurrence of DNB in the evaporator tubes for exposures up to 8,000 hours. Verification of on-line sodium ion measurement in the sub-ppm range was provided by experiments which were necessary to solidify the CRBRP chemistry specifications. CRBRP water specifications were demonstrated in a prototypical experiment.

- 1327 (GEFR--00466) FEASIBILITY OF ACOUSTIC LEAK LOCATION FOR WTD DUPLEX-TUBE STEAM GENERATORS. Gaubatz, D.C.; Greene, D.A.; Magee, P.M.; McMurtrie, K.R. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jul 1979. Contract EY-76-C-03-0893-030. 115p. AT.

It has been demonstrated that leak localization is feasible at all locations within the two Westinghouse Tampa Division (WTD) duplex-tube steam generator designs. All of the WTD requirements for leak location can be achieved. The primary WTD design requirement is that a small leak of helium gas into the shell-side sodium be located to within a cluster of seven tubes in less than four hours after hot standby conditions are achieved. This requirement can be met or exceeded everywhere in the steam generator, provided the acoustic sensors (accelerometers) are pre-mounted on the steam generator vessel. It was found to be necessary to simulate experimentally those regions of the WTD steam generator which differ significantly from the vessel wall-sodium annulus-shroud configuration of the Clinch River steam generators. Of particular concern were the convoluted bellows section and the inlet/outlet regions. Feasibility of leak location was experimentally demonstrated for both regions. The feasibility study summarized in this report consisted of twelve tasks: six areas of analysis and six experimental investigations of geometric regions, or problems, associated with the specific WTD Steam Generator designs.

- 1328 (GEFR--00468) GE-LOA-2 TASKS. 1979 ANNUAL REPORT, OCTOBER 1978-JUNE 1979. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jul 1979. Contract EY-76-C-03-0893-007. 130p. AT.

Progress in studies on safe shutdown reliability for large LMFBR type reactors is summarized under the following subtask headings: inherently safe core design, fuel pin accident transients, and fuel pin behavior modeling for TOP conditions.

- 1329 (GEFR--10028-62) CORE ENGINEERING SIXTY-SECOND QUARTERLY REPORT, FEBRUARY--APRIL

1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). May 1977. Contract EY-76-C-03-0893-011. 52p. AT.
Progress is summarized in the areas of LMFBR fuel assembly engineering and core component design analysis. (DG)
- 1330 (GEFR--10028-63) CORE ENGINEERING. SIXTY-THIRD QUARTERLY REPORT, MAY--JULY 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Aug 1977. Contract EY-76-C-03-0893-011. 41p. AT.
Progress is summarized in LMFBR fuel assembly engineering and core component design analysis. (DG)
- 1331 (GEFR--10028-64) CORE ENGINEERING. SIXTY-FOURTH QUARTERLY REPORT, AUGUST--OCTOBER 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Nov 1977. 45p. AT.
Information is presented concerning LMFBR fuel assembly engineering and core component design analysis.
- 1332 (GEFR--10028-65) CORE ENGINEERING. SIXTY-FIFTH QUARTERLY REPORT, NOVEMBER 1977--JANUARY 1978. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Feb 1978. Contract EY-76-C-03-0893-031. 27p. AT.
Progress is described. Effective October 1, 1976, the Grid-Spacer Driver Assembly and Wire-Wrap Assembly Engineering Programs were combined into a single task entitled "Fuel Assembly Engineering." Similarly, the Core Component Structural Design Criteria and Probabilistic Design Criteria were combined into a single program entitled "Core Component Design and Analysis."
- 1333 (GEFR--10028-66) CORE ENGINEERING. SIXTY-SIXTH QUARTERLY REPORT, FEBRUARY 1978--APRIL 1978. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). May 1978. Contract EY-76-C-03-0893-031. 30p. AT.
Information is presented concerning the fabrication of grid spacers and test assemblies, and classification of core components.
- 1334 (GEFR--10028-67) CORE ENGINEERING. SIXTY-SEVENTH QUARTERLY REPORT, MAY-JULY 1978. (General Electric Co., Sunnyvale, CA (USA)). Aug 1978. Contract EY-76-C-03-0893-031. 28p. AT.
Progress is described for tasks related to the grid-spaced drive assembly, wire-wrap assembly engineering, and core component design analysis.
- 1335 (GEFR--10028-68) CORE ENGINEERING. SIXTY-EIGHTH QUARTERLY REPORT, AUGUST--OCTOBER 1978. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Nov 1978. Contract EY-76-C-03-0893-031. 47p. AT.
Progress is summarized in LMFBR fuel assembly engineering and core component design analysis.
- 1336 (GEFR--10028-69) CORE ENGINEERING. SIXTY-NINTH QUARTERLY REPORT, NOVEMBER 1978--JANUARY 1979. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Feb 1979. Contract EY-76-C-03-0893-
031. 42p. AT.
Progress in LMFBR core engineering is reported under the following headings: fuel assembly engineering, and core component design analysis.
- 1337 (GEFR--13771-22) CRITICAL EXPERIMENTS AND ANALYSIS TWENTY-SECOND QUARTERLY REPORT, JANUARY--MARCH 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Apr 1977. Contract EY-76-C-03-0893-005. 77p. AT.
Current activities and technical progress for the period January through March 1977 are reported for two subtasks. Subtask A covers the technical direction for critical experiments and analysis as applicable to LMFBR commercial plants and the Clinch River Breeder Reactor Plant Benchmark Critical Experiments in ZPPR Assembly 4 and the CRBRP EMC. Subtask B is directed towards the analysis of advanced fuels critical experiments that can provide a consistent data base for testing the computational tools, nuclear analysis techniques and basic nuclear data which are used in the LMFBR design effort, and for evaluating the nuclear performance and safety characteristics of advanced fuel materials.
- 1338 (GEFR--13771-23) CRITICAL EXPERIMENTS AND ANALYSIS. TWENTY-THIRD QUARTERLY REPORT, APRIL--JUNE 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jul 1977. Contract EY-76-C-03-0893-005. 74p. AT.
Subtask A covers the technical direction for critical experiments and analysis as applicable to LMFBR commercial plants and the Clinch River Breeder Reactor Plant Benchmark Critical Experiments in ZPPR Assembly 4 and the CRBRP EMC. Subtask B is directed towards the analysis of advanced fuels critical experiments that can provide a consistent data base for testing the computational tools, nuclear analysis techniques and basic nuclear data which are used in the LMFBR design effort, and for evaluating the nuclear performance and safety characteristics of advanced fuel materials.
- 1339 (GEFR--13771-24) CRITICAL EXPERIMENTS AND ANALYSIS TWENTY-FOURTH QUARTERLY REPORT, JULY--SEPTEMBER 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Oct 1977. Contract EY-76-C-03-0893-005. 132p. AT.
Current activities and technical progress for the period July through September 1977 are reported for two subtasks. Subtask A covers the technical direction for critical experiments and analysis as applicable to LMFBR commercial plants and the Clinch River Breeder Reactor Plant Benchmark Critical Experiments in ZPPR Assembly 4 and the CRBRP EMC. Subtask B is directed towards the analysis of advanced fuels critical experiments that can provide a consistent data base for testing the computational tools, nuclear analysis techniques and basic nuclear data which are used in the LMFBR design effort, and for evaluating the nuclear performance and safety characteristics of advanced fuel materials.
- 1340 (GEFR--13771-25) CRITICAL EXPERIMENTS AND ANALYSIS. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). 1977. Contract EY-76-C-03-0893-031. 64p. AT.
Current activities and technical progress are reported for two subtasks. Subtask A covers the technical direction for critical

experiments and analysis as applicable to LMFBR commercial plants and the Clinch River Breeder Reactor Plant Benchmark Critical Experiments in ZPPR Assembly 4 and the CRBRP EMC. Subtask B is directed towards the analysis of advanced fuels critical experiments that can provide a consistent data base for testing the computational tools, nuclear analysis techniques and basic nuclear data which are used in the LMFBR design effort, and for evaluating the nuclear performance and safety characteristics of advanced fuel materials.

- 1341 (GEFR--13771-26) CRITICAL EXPERIMENTS AND ANALYSIS. TWENTY-SIXTH QUARTERLY REPORT, JANUARY--MARCH 1978. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Apr 1978. Contract EY-76-C-03-0893-031. 56p. Dep. NTIS, PC A04/MF A01.

Current activities and technical progress are reported for two subtasks. Subtask A covers the technical direction for critical experiments and analysis as applicable to LMFBR commercial plants and the Clinch River Breeder Reactor Plant Benchmark Critical Experiments in ZPPR Assembly 4 and the CRBRP EMC. Subtask B is directed towards the analysis of advanced fuels critical experiments that can provide a consistent data base for testing the computational tools, nuclear analysis techniques, and basic nuclear data which are used in the LMFBR design effort, and for evaluating the nuclear performance and safety characteristics of advanced fuel materials.

- 1342 (GEFR--13771-27) CRITICAL EXPERIMENTS AND ANALYSIS. TWENTY-SEVENTH QUARTERLY REPORT, APRIL--JUNE 1978. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jul 1978. Contract EY-76-C-03-0893-031. 61p. AT.

Current activities and technical progress for the period April through June 1978 are reported for two subtasks. Subtask A covers the technical direction for critical experiments and analysis as applicable to LMFBR commercial plants and the Clinch River Breeder Reactor Plant Benchmark Critical Experiments in ZPPR Assembly 4 and the CRBRP EMC. Subtask B is directed towards the analysis of advanced fuels critical experiments that can provide a consistent data base for testing the computational tools, nuclear analysis techniques and basic nuclear data which are used in the LMFBR design effort, and for evaluating the nuclear performance and safety characteristics of advanced fuel materials.

- 1343 (GEFR--13771-28) CRITICAL EXPERIMENTS AND ANALYSIS. TWENTY-EIGHTH QUARTERLY REPORT, JULY--SEPTEMBER 1978. (General Electric Co., Sunnyvale, CA (USA)). Oct 1978. Contract EY-76-C-03-0893-031. 84p. AT.

Current activities and technical progress are reported for the technical direction for critical experiments and analysis as applicable to LMFBR commercial plants and the Clinch River Breeder Reactor Plant Benchmark Critical Experiments in ZPPR Assembly 4 and the CRBRP EMC.

- 1344 (GEFR--13771-29) CRITICAL EXPERIMENTS AND ANALYSIS. TWENTY-NINTH QUARTERLY REPORT, OCTOBER--DECEMBER 1978. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jan 1979. Contract EY-76-C-03-0893-031. 63p. AT.

Current activities and technical progress for the period October through December 1978 are reported. This report covers the technical

direction for critical experiments and analysis as applicable to LMFBR commercial plants and the Clinch River Breeder Reactor Plant Benchmark Critical Experiments in ZPPR Assembly 4 and the CRBRP EMC.

- 1345 (GEFR--13771-30) CRITICAL EXPERIMENTS AND ANALYSIS. THIRTIETH QUARTERLY REPORT, JANUARY--MARCH 1979. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Apr 1979. Contract EY-76-C-03-0893-031. 57p. AT.

This report is the thirtieth in a series of quarterly reports published under the LMFBR Critical Experiments and Analysis Program. Current activities and technical progress for the period January through March 1979 are reported herein. This report covers the technical direction for critical experiments and analysis as applicable to LMFBR commercial plants and the Clinch River Breeder Reactor Plant.

- 1346 (GEFR--13771-31) CRITICAL EXPERIMENTS AND ANALYSIS. THIRTY-FIRST QUARTERLY REPORT, APRIL-JUNE 1979. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jul 1979. Contract EY-76-C-03-0893-031. 43p. Dep. NTIS, PC A02/MF A01.

Current activities and technical progress for the period April through June 1979 are reported herein. This report covers the technical direction for critical experiments and analysis as applicable to LMFBR commercial plants and the Clinch River Breeder Reactor Plant.

- 1347 (GEFR--13825-20-22) INSTRUMENTATION DEVELOPMENT TWENTY-SECOND QUARTERLY REPORT, OCTOBER 1976--JUNE 1977. Imhoff, D.H.; Craig, C.N. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jul 1977. Contract EY-76-C-03-0893-009. 14p. AT.

The report summarizes progress in the development of instrumentation for surveillance of the core and in-vessel components of liquid metal cooled fast breeder reactors. Current development efforts focus on high temperature neutron detectors and advanced signal transmission methods.

- 1348 (GEFR--13923-19) FUEL ROD MECHANICS AND FAILURE ANALYSIS. NINETEENTH QUARTERLY REPORT, FEBRUARY--APRIL 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). May 1977. Contract EY-76-C-03-0893-002. 46p. AT.

The objective of the work reported under this program is to model fuel rod performance and cladding failure under hypothetical transient conditions, and to validate these models using data from both integral fuel rod experiments and differential in-reactor and out-of-reactor experiments. Progress is reported under the following headings: experimental analysis and code validation, FTR and CRBR supporting analyses, and evaluation-adaptation of existing codes.

- 1349 (GEFR--13923-20) FUEL ROD MECHANICS AND FAILURE ANALYSIS. TWENTIETH QUARTERLY REPORT, MAY--JULY 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Aug 1977. Contract EY-76-C-0893-002. 61p. AT.

The objective of the work reported under this program is to model fuel rod performance and cladding failure under hypothetical transient conditions, and to validate these

models using data from both integral fuel rod experiments and differential in-reactor and out-of-reactor experiments.

- 1350 (GEFR--13923-21) FUEL ROD MECHANICS AND FAILURE ANALYSIS. TWENTY-FIRST QUARTERLY REPORT, AUGUST--OCTOBER 1977. Horst, K.M.; Craig, C.N. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Nov 1977. Contract EY-76-C-03-0893-002. 52p. AT.

The objective of the work reported under this program is to model fuel rod performance and cladding failure under hypothetical transient conditions, and to validate these models using data from both integral fuel rod experiments and differential in-reactor and out-of-reactor experiments. Progress is reported under the following headings: experiment analysis and code validation, FTR and CRBR supporting analysis, evaluation-adaptation of existing codes, fuel performance code development, code standardization and documentation, and technical management.

- 1351 (GEFR--13923-22) FUEL ROD MECHANICS AND FAILURE ANALYSIS TWENTY-SECOND QUARTERLY REPORT, NOVEMBER 1977--JANUARY 1978. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Feb 1978. Contract EY-76-C-03-0893-032. 30p. AT.

The objective of the work reported under this program is to model fuel rod performance and cladding failure under hypothetical transient conditions, and to validate these models using data from both integral fuel rod experiments and differential in-reactor and out-of-reactor experiments. Progress during November and December 1977, and January 1978 is described.

- 1352 (GEFR--13923-23) FUEL ROD MECHANICS AND FAILURE ANALYSIS. TWENTY-THIRD QUARTERLY REPORT, FEBRUARY 1978--APRIL 1978. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). May 1978. Contract EY-76-C-03-0893-032. 30p. AT.

This report is the twenty-third in a series of quarterly progress reports published under this program. The program title for the first sixteen reports was "Safety Engineering". The present title reflects more accurately the nature of the work that has been performed in the past and the work that is expected to be emphasized in the future. The objective of the work reported under this program is to model fuel rod performance and cladding failure under hypothetical transient conditions, and to validate these models using data from both integral fuel rod experiments and differential in-reactor and out-of-reactor experiments. Progress is summarized under the following headings: experiment analysis and code validation, fuel performance code development, parametric accident analysis, FTR and CRBR supporting analysis, and LOA-2 reliability assessment.

- 1353 (GEFR--13923-24) FUEL ROD MECHANICS AND FAILURE ANALYSIS. TWENTY-FOURTH QUARTERLY REPORT, MAY--JULY 1978. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Aug 1978. Contract EY-76-C-03-0893-032. 43p. AT.

The objective of the work reported is to model fuel rod performance and cladding failure under hypothetical transient conditions, and to validate these models using data from both integral fuel rod experiments and differential in-reactor and out-of-reactor experiments.

Tensile data are presented for 20% CW, type-316 stainless steel.

- 1354 (GEFR--14023-4) RISK ANALYSIS METHODS DEVELOPMENT. FOURTH QUARTERLY REPORT, OCTOBER--DECEMBER 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jan 1978. Contract EY-76-C-03-0893-032. 17p. AT.

The pilot study to identify R and D needs for energetics control during the initiating phase of a whole-core disruption accident was completed. The study showed that the LOA3 conditional failure probability goal of 0.01 given LOA2 failure is not met based on available information. R and D programs with quantitative objectives satisfying the LOA3 goal were identified, thus demonstrating the value of the LOA strategy in producing measurable objectives. Work on the LOA Integration Plan was initiated. Objectives and scope of the plan were defined. The scope of the LOA supporting analysis for this fiscal year was defined. The analysis will focus on one Level 2 product for each LOA and will use the procedure developed for the pilot study on energetics control with appropriate modifications. Objectives and scope for the LOA Integration Task Level 2 products of Risk Assessment, Cost-Benefit Analysis, and State-of-Technology Documentation were defined. The draft of the Procedural Manual for Breeder Reactor Risk Assessment, GEFR-00259(DR), was completed.

- 1355 (GEFR--14023-5) RISK ANALYSIS METHODS DEVELOPMENT. FIFTH QUARTERLY REPORT, JANUARY--MARCH 1978. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Apr 1978. Contract EY-76-C-03-0893-032. 27p. AT.

The structure of two procedures for guiding R and D planning using the line-of-assurance (LOA) approach is presented. The first procedure (LOA Integration Procedure) identifies failure dependencies between the LOAs and evaluates their effect on the LOA success probability. The second procedure (Cost-Benefit Analysis Procedure) identifies the optimum R and D choice and ranks R and D options in terms of their expected net benefit. The relationship of these procedures to the R and D planning process is also presented. The LOA Integration Procedure takes as input individual LOA definitions (given in terms of the challenge envelope for which the LOA is designed), LOA success paths, and success probability and develops corresponding event trees with consideration of challenges beyond the challenge envelope. The resulting event trees are then linked for analysis of failure dependencies resulting from causes beyond the challenge envelope, random causes within the challenge envelope, and uncertainty in the predictive tools used for evaluating the LOA performance. Probability models for LOA failure with consideration of these dependencies are presented for the four LOAs. The Cost-Benefit Analysis Procedure defines R and D program options in terms of factors describing the R and D program success criterion, test design, and test facility characterization. The procedure confines itself to economic benefits only and leaves it to the decision-maker to compare various benefits based on his own non-economic values. Separation of economic and non-economic values was accomplished by designing the procedure to provide a ranking of test facility characterization options based on the R and D program and test design which gives the maximum expected net economic benefit. Procedures for defining the R and D success

criterion and evaluating probability of the R and D program success are illustrated by examples. Cost and benefit functions are also described.

- 1356 (GEFR--14023-6/7) RISK ANALYSIS METHODS DEVELOPMENT. SIXTH--SEVENTH QUARTERLY REPORTS, APRIL--SEPTEMBER 1978. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Oct 1978. Contract EY-76-C-03-0893-032. 48p. AT.

Progress is reported for the following subtasks in risk analysis methods development for LMFBRs: Lines-of-Assurance (LOA) supporting analysis, consultation with LOA working groups, and methods and procedures development.

- 1357 (GEFR--14023-8) RISK ANALYSIS METHODS DEVELOPMENT. EIGHTH QUARTERLY REPORT, OCTOBER--DECEMBER 1978. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jan 1979. Contract EY-76-C-03-0893-032. 28p. AT.

The objectives of the program are: (1) to develop and apply methods for R and D planning in support of the Lines-of-Assurance (LOA) strategy and (2) to establish a credible Breeder Reactor (BR) risk assessment basis for evaluating R and D overall risk goal accomplishments. The report describes work on the Procedural Manual for Breeder Reactor Risk Assessment and development of a consistent probability model for single plant risk assessment.

- 1358 (GEFR--14023-9) RISK ANALYSIS METHODS DEVELOPMENT. NINTH QUARTERLY REPORT, JANUARY--MARCH 1979. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Apr 1979. Contract EY-76-C-03-0893-032. 51p. AT.

Progress in LMFBR risk analysis methods development is reported under the following headings: (1) task planning, management and liaison; (2) Lines-of-Assurance (LOA) supporting analysis; (3) consultation with LOA working groups; and (4) methods and procedures development.

- 1359 (GEFR--14023-10) RISK ANALYSIS METHODS DEVELOPMENT. TENTH QUARTERLY REPORT, APRIL--JUNE 1979. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jul 1979. Contract EY-76-C-03-0893-032. 32p. Participants in the LMFBR Program. Others request from TIC.

Progress in LMFBR risk analysis methods development is reported for the following subtasks: task planning, management and liaison; risk analysis applications; LOA (Lines of Assurance) supporting analysis; consultation with LOA working groups; and methods and procedures development.

- 1360 (GEFR--14029-12) STEAM GENERATOR MATERIALS ENGINEERING TWELFTH QUARTERLY REPORT, APRIL--JUNE 1977. Challenger, K.D. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jul 1977. Contract EY-76-C-03-0893-018. 42p. AT.

The primary objective of the work reported here is to characterize materials, mainly 2 1/4 Cr-1 Mo and stainless steels, to assure that satisfactory materials compatibility is achieved with expected LMFBR steam generator environments. Data are presented on decarburization phenomena, steam corrosion, and fracture properties.

- 1361 (GEFR--14029-13) STEAM GENERATOR MATERIALS ENGINEERING THIRTEENTH QUARTERLY REPORT, JULY--SEPTEMBER 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Oct 1977. Contract EY-76-C-03-0893-018. 35p. AT.

The primary objective of the work reported is to characterize materials, mainly 2 1/4 Cr-1 Mo and stainless steels, to assure that satisfactory materials compatibility is achieved with expected LMFBR steam generator environments.

- 1362 (GEFR--14031-9) CORE RESTRAINT ENGINEERING. NINTH QUARTERLY REPORT, AUGUST--OCTOBER 1976. Horst, K.M.; Cragi, C.N. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Nov 1976. Contract EY-76-C-03-0893-023. 14p. AT.

Progress is described for subtasks related to refueling force evaluation and test, instrumented simulated fuel assemblies, core restraint transient analysis, in-reactor monitoring, advanced core restraint studies, and seismic evaluation.

- 1363 (GEFR--14031-10) CORE RESTRAINT ENGINEERING. TENTH QUARTERLY REPORT, NOVEMBER--DECEMBER 1976 AND JANUARY 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Feb 1977. Contract EY-76-C-03-0893-023. 17p. AT.

The design guidelines and the corresponding technology required to design LMFBR core restraint systems and to support applicable licensing processes are discussed. Progress is described for subtasks related to refueling force evaluation and test, instrumented simulated fuel assemblies, core restraint transient analysis, in-reactor monitoring, advanced core restraint studies, and seismic evaluation.

- 1364 (GEFR--14031-11) CORE RESTRAINT ENGINEERING. ELEVENTH QUARTERLY REPORT, FEBRUARY--APRIL 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). May 1977. Contract EY-76-C-03-0893-023. 35p. Dep. NTIS, PC A03/MF A01.

The objective of the work reported is to develop, in conjunction with other members of the National Core Restraint Development Program, the design guidelines and the corresponding technology required to design LMFBR core restraint systems and to support applicable licensing processes. Progress during the reporting period is described for subtasks related to refueling force evaluation and test, instrumented simulated fuel assemblies, core restraint transient analysis, in-reactor monitoring, advanced core restraint studies, and seismic evaluation.

- 1365 (GEFR--14031-12) CORE RESTRAINT ENGINEERING. TWELFTH QUARTERLY REPORT, MAY--JUNE 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Aug 1977. Contract EY-76-C-03-0893-023. 25p. AT.

The design guidelines and the corresponding technology required to design LMFBR core restraint systems and to support applicable licensing processes are discussed. Progress is described for subtasks related to refueling force evaluation and test, instrumented simulated fuel assemblies, core restraint transient analysis, in-reactor monitoring, advanced core restraint studies, and seismic evaluation.

1366 (GEFR--14031-13) CORE RESTRAINT ENGINEERING. THIRTEENTH QUARTERLY REPORT, AUGUST--OCTOBER 1977. Horst, K.M.; Craig, C.N. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Nov 1977. Contract EY-76-C-03-0893-023. 24p. AT.

Progress is described for subtasks related to refueling force evaluation and test, instrumented simulated fuel assemblies, core restraint transient analysis, in-reactor monitoring, advanced core restraint studies, and seismic evaluation.

1367 (GEFR--14031-14) CORE RESTRAINT ENGINEERING. FOURTEENTH QUARTERLY REPORT, NOVEMBER 1977--JANUARY 1978. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Feb 1978. Contract EY-76-C-03-0893-031. 17p. AT.

The objective of the work reported is to develop, in conjunction with other members of the National Core Restraint Development Program, the design guidelines and the corresponding technology acquired to design LMFBR core restraint systems and to support applicable licensing processes. Progress during the reporting period is described for subtasks related to refueling force evaluation and test, instrumented simulated fuel assemblies, core restraint transient analysis, in-reactor monitoring, advanced core restraint studies, and seismic evaluation.

1368 (GEFR--14031-15) CORE RESTRAINT ENGINEERING. FIFTEENTH QUARTERLY REPORT, FEBRUARY--APRIL 1978. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). May 1978. Contract EY-76-C-03-0893-031. 19p. AT.

Progress is summarized in the following areas: (1) National Core Restraint Development Program activities, (2) refueling force evaluation and test, (3) core restraint transient analysis, (4) in-reactor monitoring, (5) seismic evaluation, and (6) core restraint system design guidelines. (DG)

1369 (GEFR--14031-16) CORE RESTRAINT ENGINEERING. SIXTEENTH QUARTERLY REPORT, MAY--JULY 1978. Horst, K.M.; Craig, C.N. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Aug 1978. Contract EY-76-C-03-0893-031. 20p. AT.

Progress during the reporting period is described for core mechanics and configuration behavior.

1370 (GEFR--14031-17) CORE RESTRAINT ENGINEERING. SEVENTEENTH QUARTERLY REPORT, AUGUST--OCTOBER 1978. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Nov 1978. Contract EY-76-C-03-0893-031. 37p. AT.

The overall objective of the program is to develop, in conjunction with other members of the National Core Mechanics and Configuration Performance Program, the design guidelines and the corresponding technology required to design LMFBR core restraint systems and to support applicable licensing processes.

1371 (GEFR--14034-11) RADIOLOGICAL ASSESSMENT MODELS. ELEVENTH QUARTERLY REPORT, MARCH--MAY 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jun 1977. Contract EY-76-C-03-0893-015. 54p. AT.

The objectives of the program are to develop models based on analyses and experiments to determine: (1) the amount and the characteristics of radiological materials discharged energetically from the core, transported to the cover-gas space and available for leakage from the reactor vessel, and (2) the long-term radiological source resulting from a core melt-through as a result of hypothetical core disruptive accidents in liquid metal fast breeder reactors. During this reporting period, the experiments to simulate and examine the characteristics of rapidly expanding saturated fuel by rapidly expanding mixtures of hot saturated water and steam have been completed. The test data have been compared with analytical results. The fuel-vapor transport model (BUBBLE I), which was reported in the tenth quarterly report, was extended to account for fuel solidification during cool down and to include a scheme to treat condensation of fuel as an aerosol source and evaluate its size distribution. A parametric study was made to determine the effects of the number of bubbles and the bubble surface emissivity on the bubble characteristics and mass of fuel transported to the cover-gas space.

1372 (GEFR--14034-13) RADIOLOGICAL ASSESSMENT MODELS. THIRTEENTH QUARTERLY REPORT, SEPTEMBER--NOVEMBER 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Dec 1977. Contract EY-76-C-03-0893-015. 33p. AT.

The objective of the program is to develop models based upon analysis and experiment for determining: (a) the radiological source term resulting from the energetic phase of a hypothetical core disruptive accident, (b) the long duration radiological source term resulting from a core melt-through, and (c) the attenuation of radiological source terms within secondary containment. The discharge of fuel into a sodium pool as a result of an HCDA depends largely upon fuel pressure, heat transfer to structure and flow rates through channels. During this quarter, equations of state for fuel were examined and a theory was developed for evaluating choked flow rates. The theory compared favorably with test data. The BUBBLES code developed to calculate the collapse of vapor bubbles containing noncondensable gas was used to examine the effects of variations in the amount of gas. The code was improved and calculations were continued in order to determine bubble heat transfer parameters.

1373 (GEFR--14034-14) RADIOLOGICAL ASSESSMENT MODELS FOURTEENTH QUARTERLY REPORT, DECEMBER 1977--FEBRUARY 1978. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Mar 1978. Contract EY-76-C-03-0893-032. 16p. AT.

The report documents work performed to develop models based on analyses and experiments which determine: (a) radiological source terms resulting from hypothetical, energetic and melt-through core disruptive accident scenarios, and (b) radiological source term attenuation within containment. The BUBBLES code being developed to evaluate potential radiological source terms resulting from energetic hypothetical core disruptive accidents has been used to calculate the collapse of water vapor bubbles containing noncondensable gas. The results have been compared with test data to evaluate heat transfer parameters. Compared with the test data, the BUBBLES code provides conservative results over the range of test Peclet numbers.

Similar results are expected for sodium vapor bubbles.

- 1374 (GEFR--14034-15) RADIOLOGICAL ASSESSMENT MODELS. FIFTEENTH QUARTERLY REPORT, MARCH--MAY 1978. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jun 1978. Contract EY-76-C-03-0893-032. 21p. AT.

The objective of the program is to develop models based on analyses and experiments which determine: (a) radiological source terms resulting from hypothetical, energetic and melt-through core disruptive accident scenarios, and (b) radiological source term attenuation within containment. The BUBBLES code being developed to evaluate attenuation of potential radiological sources resulting from energetic hypothetical core disruptive accidents was extended to treat condensation of sodium vapor bubbles. Parametric calculations were performed for a representative hypothetical accident in order to evaluate the effects of noncondensable gas and surface area (number of bubbles) on condensation rates. The results indicate that for dispersed discharge from the core there is a large potential for source term attenuation even when large amounts of noncondensable gas are present.

- 1375 (GEFR--14034-16) RADIOLOGICAL ASSESSMENT MODELS. SIXTEENTH QUARTERLY REPORT, JUNE--AUGUST 1978. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Sep 1978. Contract EY-76-C-03-0893-032. 25p. AT.

This report, the sixteenth in a series of quarterly reports published under the Radiological Assessments program, documents work performed from June 1978 to August 1978. The objective of the program is to develop models based on analyses and experiments which determine: (a) radiological source terms resulting from hypothetical, energetic and melt-through core disruptive accident scenarios, and (b) radiological source term attenuation within containment. Test data pertaining to fuel and fission product release from burning and boiling sodium pools were examined to determine the mechanisms which control such releases. The data were found to be inadequate to determine conclusively the processes involved. A theoretical investigation is in progress to identify controlling parameters, develop correlations between them, and define experiments. Sensitivity studies over the range of uncertainty in decontamination factors indicate the risk is acceptable for the extremely low probability accidents considered in lines of assurance three and four. It is expected the risk could be significantly reduced if more test data were available.

- 1376 (GEFR--14038-11) ADVANCED SAFETY ANALYSIS. ELEVENTH QUARTERLY REPORT, MARCH--MAY 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jun 1977. Contract EY-76-C-03-0893-019. 126p. AT.

Information on LMFBR safety is presented concerning engineering analysis of post-accident heat removal; inherently safe core design; fuel pin accident transients; safe shutdown reliability for large LMFBR; and experimental fuel rod fabrication.

- 1377 (GEFR--14038-12) ADVANCED SAFETY ANALYSIS. TWELFTH QUARTERLY REPORT, JUNE--AUGUST 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor

Dept.). Sep 1977. Contract EY-76-C-03-0893-019. 104p. AT.

Progress is summarized in the following study areas: engineering analysis of post-accident heat removal, inherently safe core design, fuel-pin accident transients, safe shutdown reliability for large LMFBRs, and experimental fuel rod fabrication.

- 1378 (GEFR--14038-13) ADVANCED SAFETY ANALYSIS THIRTEENTH QUARTERLY REPORT, SEPTEMBER--NOVEMBER 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Dec 1977. Contract EY-76-C-03-0893-032. 101p. AT.

Progress is summarized in the following study areas: accident debris analysis, inherently safe core design, fuel pin accident transients, safe shutdown reliability for large breeder reactors, and experimental fuel rod fabrication.

- 1379 (GEFR--14038-15) ADVANCED SAFETY ANALYSIS. FIFTEENTH QUARTERLY REPORT, MARCH--MAY 1978. (General Electric Co., San Jose, CA (USA). Boiling Water Reactor Systems Dept.). Jun 1978. Contract EY-76-C-03-0893-032. 59p. AT.

Progress is summarized in the following areas of LMFBR safety research: accident debris analysis, inherently safe core design, fuel pin accident transients program, safe shutdown reliability for large breeder reactors, and experimental fuel rod fabrication.

- 1380 (GEFR--14073-3) NUCLEAR DATA AND METHODS DEVELOPMENT AND APPLICATION TO LMFBRs. THIRD ANNUAL PROGRESS REPORT, FISCAL YEAR 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). 1977. Contract EY-76-C-03-0893-021. 49p. AT.

This report is the third in a series of progress reports published under the Cooperative Nuclear Data and Methods Development Program. The objective of the work reported under this program is to support cooperative evaluations of nuclear data and analytical design tools, to expedite implementation of these in-reactor design applications, and to perform specific analytical physics tasks required to support the Fast Breeder Reactor Program.

- 1381 (GEFR--14073-4) NUCLEAR DATA AND METHODS DEVELOPMENT AND APPLICATION TO LMFBRs. FOURTH ANNUAL PROGRESS REPORT, FISCAL YEAR 1978. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 1978. Contract EY-76-C-03-0893-031. 48p. AT.

The fast critical assemblies ZPR-6-7 and ZPR-9-31 were analyzed as part of Phase II testing of the preliminary Version V of ENDF/B. The analysis of the large core sodium-voiding benchmark has been completed. The conclusions based on these studies are reported. The radiological hazards from the actinide wastes discharged from a number of alternate fast reactor systems have been determined. It is recommended that further actinide recycle efforts include the investigation of a special purpose actinide-burner reactor. A systematic analysis of neutronics methods for large heterogeneous cores is being carried out.

- 1382 (GEFR--14074-5) COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT FIFTH QUARTERLY REPORT, OCTOBER--DECEMBER 1976. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jan 1977. Contract EY-76-C-

03-0893-021. 32p. AT.

Progress is reported in the cooperative evaluation of nuclear data and analytical design tools for LMFBRs.

- 1383 (GEFR--14074-6) COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT SIXTH QUARTERLY REPORT, JANUARY--MARCH 1977. (Analysis and Computer Systems, Inc., Burlington, MA (USA)). Apr 1977. Contract EY-76-C-03-0893-021. 31p. AT.

Progress is reported in the cooperative evaluation of nuclear data and analytical design tools for LMFBRs.

- 1384 (GEFR--14074-7) COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT SEVENTH QUARTERLY REPORT, APRIL--JUNE 1977. Horst, K.M.; Craig, C.N. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jul 1977. Contract EY-76-C-03-0893-021. 38p. AT.

The report presented in the seventh in a series of quarterly reports published under the Cooperative Nuclear Data and Methods Development Program. The objective of the work reported under this program is to support cooperative evaluations of nuclear data and analytical design tools, to expedite implementation of these in-reactor design applications, and to perform specific analytical physics tasks required to support the Fast Breeder Reactor Program including an evaluation of the nuclear feasibility of significantly impacting the waste disposal problem by utilizing actinide recycle.

- 1385 (GEFR--14074-8) COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT. EIGHTH QUARTERLY REPORT, OCTOBER--DECEMBER 1977. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jan 1978. Contract EY-76-C-03-0893-031. 44p. AT.

The Phase I Reviews of U-238, Cf-249, Cf-250 and Cf-251 as evaluated for Version V of ENDF/B have been completed. The recommended Pu-239 data for ENDF/B-V has been tested for a fast critical assembly to determine the effects of the changes with respect to ENDF/B-IV. Calculations for several fast CSEWG benchmarks have been performed using ENDF/B-IV data and the latest version of the data processing code TDOWN. The second LCCEWG benchmark, which is the sodium voiding of a large homogeneous fast reactor, has been analyzed. The results for the 21- and 4-energy group calculations are reported. A survey of the literature on actinide partitioning and actinide burnup has been carried out.

- 1386 (GEFR--14074-9) COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT. NINTH QUARTERLY REPORT, JANUARY--MARCH 1978. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Apr 1978. Contract EY-76-C-03-0893-031. 47p. AT.

The objective of the work reported is to support cooperative evaluations of nuclear data and analytical design tools to expedite their implementation in reactor design applications and to perform specific analytical physics tasks required in the fast breeder reactor program. These tasks include an evaluation of the nuclear feasibility of significantly impacting the waste disposal problem by utilizing actinide recycle.

- 1387 (GEFR--14074-10) COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT. TENTH QUARTERLY REPORT, APRIL--JUNE 1978. (General Electric

Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jul 1978. Contract EY-76-C-03-0893-031. 39p. AT.

The objective of the work reported is to support cooperative evaluations of nuclear data and analytical design tools to expedite their implementation in reactor design applications and to perform specific analytical physics tasks required in the fast breeder reactor program. These tasks include an evaluation of the nuclear feasibility of significantly impacting the waste disposal problem by utilizing actinide recycle.

- 1388 (GEFR--14074-11) COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT. ELEVENTH QUARTERLY REPORT, JULY--DECEMBER 1978. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jan 1979. Contract EY-76-C-03-0893-031. 53p. AT.

The ^{239}Pu to ^{235}U fission cross section ratios were reevaluated in the energy range 0.2 to 12 MeV. The reevaluated ^{239}Pu to ^{235}U fission cross section ratios were used to obtain the ^{239}Pu fission cross sections for ENDF/B-V. To conform with DOE requirements on the release of computer codes, an effort is underway to convert TDOWN-III to the CCC standard interface format for input/output processing. A more accurate and efficient method for generating the transport kernels used in TDOWN has been developed. It is based on the exact solution of the transport equation in slab geometry, consistent with a flat source assumption. The solution is extended to cylindrical geometry using a slicing technique. Uncertainties in the cross section data for the higher plutonium and americium isotopes were investigated to determine their impact on the important LMFBR design parameters. The results of this investigation indicated that the design calculations are sensitive to the uncertainties in the capture and fission data for ^{240}Pu and ^{241}Pu , and for long-term design applications these uncertainties should be reduced to less than about 10%. The uncertainties in the data for ^{242}Pu and for the americium isotopes have only a small impact on the fast reactor design studies but may be important in the development of schemes for the management of actinide waste discharge and for plutonium burnout. The treatment of control rod heterogeneity effects with two-dimensional modeling was investigated. The results indicate that reasonable k-effectives can be computed for various control in/control out configurations in (X,Y) geometry using axial bucklings obtained from a single (R,Z) problem with central control rod in and with control out annuli representing the remaining control rod positions.

- 1389 (GEFR--14074-12) COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT. TWELFTH QUARTERLY REPORT, JANUARY--MARCH 1979. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Apr 1979. Contract EY-76-C-03-0893-031. 38p. AT.

The GE-ARSD requirements for an ENDF/B-V multigroup library were specified and forwarded to ORNL. An effort is underway to convert TDOWN-III to the CCC standard interface format for input/output processing. A more accurate and efficient method for generating transport kernels was incorporated into TDOWN-III. The test calculations verified that excellent results are obtained at a reduced cost. Two actinide burner reactors were analyzed by use of one-dimensional diffusion theory and the point burnup code ORIGEN. These include a liquid-fuel spherical reactor with a high actinide burnup rate and an actinide-fueled fast reactor similar to an LMFBR. Reasonable k-

effectives can be computed with diffusion theory for various control-in/control-out/control partially configurations in (X,Y) geometry by use of axial bucklings obtained from (R,Z) problems with the central control fully or partially inserted and with control-out annuli representing the remaining control rod positions. These conclusions were verified by three-dimensional calculations. However, differences between (R,Z) and (X,Y) transport theory calculations were larger than expected for the same procedures. These differences may be due to the ray-effect in (X,Y) geometry, requiring the use of higher order $S_{\text{sub } n}$. An investigation of the transient behavior of homogeneous and heterogeneous LMFBR designs using the two-dimensional space-kinetics code FX2-TM was begun. 5 figures, 10 tables.

- 1390 (GEFR--14074-13) COOPERATIVE NUCLEAR DATA AND METHODS DEVELOPMENT. THIRTEENTH QUARTERLY REPORT, APRIL-JUNE, 1979. Protsik, R. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jul 1979. Contract EY-76-C-03-0893-031. 21p. AT.

The objective of the Cooperative Nuclear Data and Methods Program is the development and working implementation of nuclear data and methods relevant to the design and analysis of fast breeder reactors. The specific objectives of the four individual subtasks are summarized as follows: Cooperative Nuclear Data Development - To participate in the activities of the Cross Section Evaluation Working Group (CSEWG) to evaluate the nuclear data which is relevant to the design of fast breeder reactors. Cooperative Methods Development - To participate in the development of nuclear design methods including code validation in the functional areas of data processing, the analysis of critical experiments, and the design of large power reactors. Actinide Recycle Analysis - To evaluate actinide recycle in LMFBRs and the eventual use of Pu-burners as a waste management option. Full Scale LMFBR Analysis - To evaluate methods and quantify approximations for use in the analysis of heterogeneous cores.

- 1391 (GEFR-SP--010) OUT-OF-PILE EXPERIMENTS PERFORMED IN THE U.S. FUEL CLADDING CHEMICAL INTERACTION (FCCI) PROGRAM. Adamson, M.G. (General Electric Co., Pleasanton, CA (USA). Vallecitos Nuclear Center). 1977. Contract EY-76-C-03-0893-014. 69p. (CONF-770215--2). Dep. NTIS, PC A04/MF A01.

From Technical committee meeting on fuel and cladding interaction; Tokyo, Japan (21 Feb 1977).

Since 1972 a variety of out-of-pile experiments have been performed as part of the US National Fuel-Cladding Chemical Interaction (FCCI) Program. In the present paper results from these experiments are presented together with descriptions of many of the experimental techniques employed to obtain them. Although the main emphasis of the paper is on experiments designed to characterize FCCI with Type-316SS cladding, considerable attention is also paid to the following FCCI-related topics: thermodynamics of and phase equilibria in mixed oxide fuel and fission product compounds, fission product and cladding component thermo-transport, and chemical behavior of candidate oxygen-absorber materials (buffer/getters). Detailed interpretations of these results in terms of FCCI mechanisms are presented in a companion paper.

CLADDING CHEMICAL INTERACTION: US INTERPRETATION. Adamson, M.G. (General Electric Co., Pleasanton, CA (USA). Vallecitos Nuclear Center). 1977. Contract EY-76-C-03-0893-014. 47p. (CONF-770215--1). Dep. NTIS, PC A03/MF A01.

From Technical committee meeting on fuel and cladding interaction; Tokyo, Japan (21 Feb 1977).

Proposed mechanisms of fuel-cladding chemical interaction (FCCI) in LMFBR fuel pins are reviewed and examined in terms of in-pile and out-of-pile data. From this examination several factors are identified which may govern the occurrence of localized deep intergranular penetrations of Type-316SS cladding. Using a plausible mechanistic hypothesis for FCCI, first steps have been taken towards developing a quantitative, physically-meaningful, mathematical method of predicting cladding wastage in operating fuel pins. Both kinetic and thermodynamic aspects of FCCI are considered in the development of this prediction method, together with a fuel chemistry model that describes the evolution of thermochemical conditions at the fuel-cladding gap. On the basis of results from recent fuel pin and laboratory tests a thermal transport mechanism has been proposed to explain the thermal gradient-induced migration of Fe, Cr, and Ni from cladding into the fuel. This mechanism involves chemical transport of the metallic cladding components (as tellurides) in liquid Cs--Fe.

- 1393 (GEFR-SP--023) POST-ACCIDENT HEAT REMOVAL IN LMFBRs. Glueckler, E.L.; Baker, L. Jr. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jun 1977. Contract EY-76-C-03-0893-019. 62p. (CONF-771120--13). Dep. NTIS, PC A04/MF A01.

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

For safety assessments of early sodium cooled fast breeder reactors (LMFBRs), extremely unlikely events such as unprotected loss of flow and transient overpower sequences have been evaluated. The prediction of the consequences of these postulated events involves investigations of decay heat dissipation and core retention. In general, these problem areas are referred to collectively as post-accident heat removal (PAHR). This paper reviews data on heat transfer, fluid mechanics, nuclear stability and material interactions associated with PAHR. Some concepts applied in early LMFBRs for consequence mitigation are also described.

- 1394 (GEFR-SP--027) RISK ALLOCATION APPROACH TO REACTOR SAFETY DESIGN AND EVALUATION. Gokcek, D.; Temme, M.I.; Derby, S.L. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). 23 Aug 1977. Contract EY-76-C-03-0893-019. 7p. (CONF-780507--20). Dep. NTIS, PC A02/MF A01.

From Probabilistic analysis of nuclear reactor safety; Los Angeles, CA, USA (8 May 1978).

The techniques of probabilistic risk assessment and risk allocation are viable and valuable tools for establishing safety reliability goals for nuclear power plants. These methods allow quantitative design goals to be derived from a risk acceptance constraint. The basic concept described in this paper can be applied to other type power generating plants with minor modifications.

EVALUATIONS OF POSTULATED CORE MELT-DOWN ACCIDENTS IN EARLY-SIZED LMFBR'S. Harless, W.H.; Talwar, B.; Glueckler, E.L. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). 26 Oct 1977. Contract EY-76-C-03-0893-019. 13p. (CONF-771108--6). Dep. NTIS, PC A02/MF A01.

From Post accident heat removal information exchange meeting; Chicago, IL, USA (2 Nov 1977).

Secondary criticality analyses have been performed for an early-sized LMFBR. The results indicate that recriticality will occur only as a result of extremely unlikely combinations of failures which lead to meltdown of a large portion of the reactor core, subsequent formation of a relatively pure fuel layer, and/or substantial vessel deformation. Design features were identified that could be used to eliminate recriticality or mitigate its effects if needed to meet safety requirements. The presently existing large uncertainties in the containment response and radiological consequences of single or multiple recriticalities may be reduced with the aid of dynamic analyses that couple neutronic, thermal, and hydrodynamic methods.

1396 (GEFR-SP--042) DESIGN, MANUFACTURING, AND TEST STATUS OF THE CRBRP HOCKEY STICK STEAM GENERATORS. Little, A.F.; Bashford, A.L. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Nov 1977. Contract EY-76-C-03-1142; EY-76-C-03-1143. 41p. (CONF-780203--3). Dep. NTIS, PC A03/MF A01.

From 1. joint U.S./Japan LMFBR steam generator seminar; Tokyo, Japan (13 Feb 1978).

Portions of document are illegible.

The history of the development of the hockey stick steam generator, currently being designed and fabricated for use in the CRBRP, is traced back to its inception in the late 1960's. Additional supporting development, which has formed the basis for the current CRBR design, is presented. The design requirements for the CRBR steam generators are outlined and the basic design, which has been evolved to meet those requirements, is described. Of particular interest to the designer are the design analysis problem areas which have been encountered and the method of attack which has been employed to resolve these problems. In particular, the method of handling the creep characteristics of 2 1/4 Cr--1 Mo, as developed by the material properties program at ORNL, is presented. Finally, the current status of the prototype and CRBRP steam generator plant unit production effort is described. The critical tube-tubesheet welding and automated MIG welding of shell sections--developed especially for the CRBRP steam generators--is discussed.

1397 (GEFR-SP--044) SELECTION OF THE DESIGN BASIS LEAK FOR LMFBR STEAM GENERATORS. Meyer, R.A.; Pfefferlen, H.C.; Roberts, J.M.; Sane, J.O. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Nov 1977. Contract EY-76-C-03-0893. 17p. (CONF-780203--2). Dep. NTIS, PC A02/MF A01.

From 1. joint U.S./Japan LMFBR steam generator seminar; Tokyo, Japan (13 Feb 1978).

Steam generator tube failure mechanisms that have been observed in experiments or that have been postulated are discussed. The DBL for CRBRP and a proposed DBL for future large LMFBR steam generators are described. Safety considerations and philosophy in selection of Design Basis Leaks (DBL) are presented. A discussion of the Large Leak Test Rig (LLTR)

Series II program support of the DBL selection for future large LMFBR steam generators is included.

1398 (GEFR-SP--045) CORROSION AND IMPURITIES DEPOSITION EFFECTS UNDER DNB CONDITIONS. Rush, D.J.; Hampton, L.V.; Spalaris, C.N.; Schmidt, D.K.; Padden, T.R. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Nov 1977. Contract EY-76-C-03-0962. 21p. (CONF-780203--4). Dep. NTIS, PC A02/MF A01.

From 1. joint U.S./Japan LMFBR steam generator seminar; Tokyo, Japan (13 Feb 1978).

A summary of two separate steam generator model test programs, including a description of the test models, test systems, control and monitoring of water chemistry, in situ nondestructive examination methods and results, post-test destructive examination methods and results, are presented. The test programs described are: the DNB Corrosion Effects test program conducted by General Electric which comprised a sodium-heated single-wall-tube test model with a 4.9 m (16 feet) active heat transfer length and a single-wall heat transfer tube with 10.1 mm inside diameter prototypic of the CRBR steam generator tubing; and the Small Steam Generator Model corrosion test program conducted by Westinghouse which comprised a sodium-heated double-wall tube with 14.5 mm inside diameter and 18.6 m (60.9 feet) sodium heated length. The objectives of these tests were to determine whether accelerated or localized damage would occur as a result of DNB operation. The two parameters varied were caustic level and heat transfer. The GE, or 'prototypic,' test was conducted at an extreme condition of the plant specifications: 9 +/- 3 ppB Na and thermal conditions corresponding to 64% plant power where the heat flux and tube wall temperature oscillations are maximum at 1.26×10^6 W/m² (0.4×10^6 Btu/hr-ft²) and 56.7°C (102°F), respectively. The Westinghouse test was conducted at a 30 ppB sodium ion content to simulate worst-case conditions under which the CRBR might be operated with the thermal conditions corresponding to the less severe 100% power condition where the heat flux is about 0.75×10^6 W/m² (0.24×10^6 Btu/hr-ft²).

1399 (GEFR-SP--046) FABRICATION INSPECTION OF TUBING, TUBESHEETS, AND TUBE-TO-TUBESHEET WELDS FOR THE CRBRP STEAM GENERATORS. Spalaris, C.N.; Durand, R.E.; McClung, R.W.; Wright, E.A. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Nov 1977. Contract EY-76-C-03-0893. 33p. (CONF-780203--5). Dep. NTIS, PC A03/MF A01.

From 1. joint U.S./Japan LMFBR steam generator seminar; Tokyo, Japan (13 Feb 1978).

The rationale for the technical details leading to the selection of tubing and tubesheet specifications and inspection techniques was based upon the desire to optimize the chances of success for the CRBRP steam generators. Fabrication processes, especially welding for the tube-to-tubesheet joints were recognized as critical. Thus, heavy emphasis was placed upon the development of reliable welding techniques as well as inspection methods using the most accurate and sensitive tools available. Accomplishments to date include: a) the procurement of the tubing and tubesheets, to the desired specifications; b) validation that the design base properties of VAR and ESR wrought products are well within the Code acceptable levels; and c) development

of welding and tube-to-tubesheet inspection methods to provide satisfactory levels of quality.

- 1400 (GEFR-SP--053) CRBR PUMP DESIGN CONCEPTS AND RESULTANT DESIGN EVOLUTION. Manners, L. (General Electric Co., Sunnyvale, CA (USA)). 1977. Contract EY-76-C-03-0893-001. 16p. AT.

US-JRDA/Japanese-PNC seminar on LMFBF components, December 1977. Portions of document are illegible.

The design, model testing, and fabrication status of the primary sodium pumps for the Clinch River Breeder Reactor are discussed.

- 1401 (GEFR-SP--062) CLINCH RIVER BREEDER REACTOR: SECONDARY CONTROL ROD SYSTEM. McKeehan, E.R.; Sim, R.G. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). 14 Sep 1977. Contract EY-76-C-03-1143. 19p. AT.

The shutdown system for the Clinch River Breeder Reactor (CRBR) includes two independent systems, a primary and a secondary system. The secondary control rod system (SCRS) is a new design which is being developed by General Electric to be diverse from the primary in order to improve overall shutdown reliability by eliminating potential common-mode failures. The status of the SCRS design, fabrication and testing activities are described. Design verification testing on the component level is largely complete. These component tests will be covered with emphasis on design impact results.

- 1402 (GEFR-SP--064) LARGE ELECTROMAGNETIC PUMPS. Kilman, G.B. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). 1976. Contract EY-76-C-03-0893-017. 24p. Dep. NTIS, PC A02/MF A01.

The development of large electromagnetic pumps for the liquid metal heat transfer systems of fission reactors has progressed for a number of years. Such pumps are now planned for fusion reactors and solar plants as well. The Einstein-Szilard (annular) pump has been selected as the preferred configuration. Some of the reasons that electromagnetic pumps may be preferred over mechanical pumps and why the annular configuration was selected are discussed. A detailed electromagnetic analysis of the annular pump, based on slug flow, is presented. The analysis is then used to explore the implications of large size and power on considerations of electromagnetic skin effect, geometric skin effect and the cylindrical geometry.

- 1403 (GEFR-SP--065) LMFBF VENTED ASSEMBLY HYDRAULIC MODELING AND TESTING. Wang, J.C.F.; Dubberley, A.E. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). 1978. Contract EY-C-03-J893-031. 7p. (CONF-780622--34). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

A fundamental uncertainty in designing or evaluating the performance of an LMFBF core composed of vented assemblies is the flow pattern of the coolant through the vents and spaces between the assemblies, as controlled by the vent hydraulic characteristics. The principal difference between a conventional and vented assembly is that a vented assembly has holes cut in the duct wall at locations above the active core region below the above-core load pads. The purpose of the venting is (i) to extend assembly life by pressurizing the core

interstitial region (spaces between the assemblies) to reduce duct stress and radial growth, and (2) to reduce the reactor coolant pressure drop by providing an additional bypass flow path to the upper plenum for the coolant after it cools the active core. The paper presents analytical and test results evaluating the vent flow discharge coefficient with respect to the system flow rates and vent geometry.

- 1404 (GEFR-SP--070) HEAT TRANSFER WITH HOCKEY-STICK STEAM GENERATOR. Moody, E.; Gabler, M.J. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). Nov 1977. Contract EY-76-C-03-1142; EY-76-C-03-1143. 29p. (CONF-780203--1). Dep. NTIS, PC A03/MF A01.

From 1. joint U.S./Japan LMFBF steam generator seminar; Tokyo, Japan (13 Feb 1978).

The hockey-stick modular design concept is a good answer to future needs for reliable, economic LMFBF steam generators. The concept was successfully demonstrated in the 3t Mwt MSG test unit; scaled up versions are currently in fabrication for CRBRP usage, and further scaling has been accomplished for PLBR applications. Design and performance characteristics are presented for the three generations of hockey-stick steam generators. The key features of the design are presented based on extensive analytical effort backed up by extensive ancillary test data. The bases for and actual performance evaluations are presented with emphasis on the CRBRP design. The design effort on these units has resulted in the development of analytical techniques that are directly applicable to steam generators for any LMFBF application. In conclusion, the hockey-stick steam generator concept has been proven to perform both thermally and hydraulically as predicted. The heat transfer characteristics are well defined, and proven analytical techniques are available as are personnel experienced in their use.

- 1405 (GEFR-SP--130) MINIMIZING UNBALANCE RESPONSE OF THE CRBRP SODIUM PUMPS. Gupta, V.K.; Marrujo, F.G. (General Electric Co., Sunnyvale, CA (USA)). Advanced Reactor Systems Dept.). Apr 1979. Contract EY-76-C-15-0003. 32p. (CONF-790802--17). Dep. NTIS, PC A03/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

The unbalance response characteristics of the vertical pumps for the Clinch River Breeder Reactor Plant are investigated. Finite-element shell and beam models representative of the pump-motor structure including the rotating assembly are developed to assess structural stiffnesses of dominant joints as well as the foundation support stiffness so as to exclude the danger of resonant excitation during normal operation. Less than four mils peak-to-peak vibration amplitude at the pump tank discharge nozzle results from just 10% frequency separation between the first rocking mode and the maximum operating speed of 1116 RPM, based on 0.5% modal damping ratio and balance quality grade of ISO/ANSI G2.5 for the rotating components: motor rotor, pump shaft, Bendix diaphragm-type flexible coupling, and centrifugal double-suction impeller. Several design options are explored for raising shaft critical speed beyond 125% of maximum operating speed.

- 1406 (GEFR-SP--133) CRITERIA FOR TOP OR

LOPI TERMINATION IN A LMFBR. Cantrell, R.T.; Peck, C.B. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 12 Jan 1979. Contract EY-76-C-03-0893-032. 11p. (CONF-790816--25). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Conservative criteria are presented which relate safety test data to successful termination of Transient Overpower (TOP) and Loss of Piping Integrity (LOPI) accidents in a LMFBR. Simulated sets of test data were generated and applied to these criteria in order to predict whether real tests will establish LOA-2 success for these events.

1407 (GEFR-SP--135) EVENT TREE ANALYSIS OF CORE DISRUPTIVE ACCIDENT CONSEQUENCES SAFETY R AND D PLANNING. Peck, C.B.; Feller, K.G. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 12 Jan 1979. Contract EY-76-C-03-0893-032. 11p. (CONF-790816--25). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

These results provide a basis for planning research and development to support breeder reactor safety Line-of-Assurance 3. Consequences were calculated for different combinations of reactor containment design features and accident events assumed to occur with a 2500 MW(t) liquid metal fast breeder reactor hypothetical core meltdown. The consequences provide a figure of merit for evaluating both containment system concepts, and research which demonstrates favorable natural processes, and limits on unfavorable processes in the release of radionuclides in the environment.

1408 (GEFR-SP--136) LOADING ERROR SIMULATIONS AND CONTROL WORTH MEASUREMENTS IN A ZPPR CRITICAL ASSEMBLY OF HETEROGENEOUS CONFIGURATION. Lu, Y.S.; Crick, S.C.; Stewart, S.L. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 1979. Contract EY-76-C-03-0893. 15p. (CONF-790602--47). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

A series of loading error simulations and control worth measurements were performed in ZPPR-8, Phase F, which was a part of the benchmark study of a demonstration-reactor-sized heterogeneous (radial parfait) LMFBR core. The results are presented of both experiments and analyses, and the significance of the observation on neutron flux redistribution due to control insertion in a heterogeneous core is analyzed.

1409 (GEFR-SP--140) PREDICTION OF COOLANT MIXING AND PERIPHERAL FLOW IN WIRE WRAPPED ROD BUNDLES. Wei, J.P. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 1979. Contract EY-76-C-03-0893. 9p. (CONF-791103--14). Dep. NTIS, PC A02/MF A01.

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

Thermal-hydraulic behavior in LMFBR wire-wrapped rod bundles was studied using the FULMIX-II code and the results compared with experimental data from the French 19-pin water flow tests.

1410 (GEFR-SP--149) APPLICATION OF CORE STRUCTURAL DESIGN GUIDELINES IN CONCEPTUAL FUEL PIN DESIGN. Patel, M.R.; Stephen, J.D. (General Electric Co., Sunnyvale, CA (USA)). 14 Mar 1979. Contract EY-76-C-03-0893-031. 10p. (CONF-790615--6). Dep. NTIS, PC A02/MF A01.

From 3. US national congress on pressure vessels and piping; San Francisco, CA, USA (25 Jun 1979).

The paper describes an application of the Draft RDT Standards F9-7, -8, and -9 to conceptual design of Fast Breeder Reactor (FBR) fuel pins. The Standards are being developed to provide guidelines for structural analysis and design of the FBR core components which have limited ductility at high fluences and are not addressed by the prevalent codes. The development is guided by a national working group sponsored by the Division of Reactor Research and Technology of the Department of Energy. The development program summarized in the paper includes establishment of design margins consistent with the test data and component performance requirements, and application of the design rules in various design activities. The application program insures that the quantities required for proper application of the design rules are available from the analysis methods and test data, and that the use of the same design rules in different analysis tools used at different stages of a component design produces consistent results. This is illustrated in the paper by application of the design rules in the analysis methods developed for conceptual and more detailed designs of an FBR fuel pin.

1411 (GEFR-SP--151) WELDING FOR THE CRBRP STEAM GENERATORS. Spalaris, C.N.; Ring, P.J.; Durand, R.E.; Wright, E.A. (General Electric Co., Sunnyvale, CA (USA); Atomic International Div., Canoga Park, CA (USA); Clinch River Breeder Reactor Plant Project Office, Oak Ridge, TN (USA)). 1979. Contract EW-76-C-15-2395;EW-76-C-15-0003. 9p. (CONF-790426--2). Dep. NTIS, PC A02/MF A01.

From International conference on welding and fabrication in the nuclear industry; London, UK (2 Apr 1979).

The rationale for selecting weld design, welding procedures and inspection methods was based upon the desire to obtain the highest reliability welds for the CRBRP steam generators. To assure the highest weld reliability, heavy emphasis was placed on the control of material cleanliness and composition substantially exceeding the requirements of the ASME Code for 2-1/4Cr-1Mo. The high tube/tubesheet weld quality was achieved through close material control, an extensive weld development program and the selection of high reliability welding equipment. Shell and nozzle weld fabrication using TIG, MIG, and submerged arc procedures are also being controlled through precise specifications, including preheat and postheat programs, together with radiography and ultrasonic inspection to ascertain the weld quality desired. Details of the tube/tubesheet welding and shell welding are described and results from the weld testing program are discussed.

1412 (GEFR-SP--159) INTERACTIONS BETWEEN THERMAL AND STRUCTURAL BEHAVIOR IN FBR FUEL PINS. Harbourn, B.L.; Patel, M.R.; Stephen, J.D. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 23 Apr 1979. Contract EY-76-C-03-0893-031. 16p. (CONF-790802--38). Dep. NTIS, PC A02/MF A01.

From International conference on structural

mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

The one-dimensional LIFE4 and the multidimensional TWOD codes form part of a system of design methods used for FBR core design. The primary purpose of LIFE4 and TWOD is to determine the capability of FBR fuel pins to attain target operating conditions and exposures while remaining within prescribed thermal and structural design limits. The codes utilize fundamentally-based models for fuel behavior with the capability to account for the interactions between phenomena.

1413 (GEFR-SP--164) METHOD OF ASSESSING THE IMPACT OF METALLURGICAL NOISE AND ITS REDUCTION IN ULTRASONIC TESTING OF AUSTENITIC WELDS.

Day, R.A. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Aug 1979. Contract EY-76-C-03-0893. 23p. (CONF-790868--1). Dep. NTIS, PC A02/MF A01.

From IEEE ultrasonics symposium; New Orleans, LA, USA (Aug 1979).

A method of reducing the ultrasonic noise, or spurious echos, often present in austenitic metals has been developed. The method has been used on specimens of Type 316 stainless steel, Alloy 800, 16-8-2 weld metal, and Inco-82 type weld metal. The method has been applied to shear wave tests and is applicable to longitudinal wave tests. Noise in these materials has been subsequently reduced relative to saw cut notches in welded samples. Signal-to-noise improvements of 8Db have been achieved in 16-8-2 weld metal, and order of magnitude improvement is obtained in the probability of large noise signals per inspection. Experiments with 1 and 1/2 megahertz inspection indicate that the improvement attributed to longitudinal wave inspection may be due to the difference in wavelength rather than wave mode.

1414 (GEFR-SP--165) ANALYSIS OF THE U.S. LARGE LEAK TEST PROGRAM SERIES I SODIUM--WATER REACTION TEST RESULTS. Sane, J.O.; Regimbal, J.J.; Fairbairn, J.A.; Meyer, R.A. (General Electric Co., Sunnyvale, CA (USA)). 1979.

Contract EY-76-C-03-0893-010. 11p. (CONF-790816--11). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

A computer code for modeling Liquid Metal Fast Breeder Reactor (LMFBR) Steam Generator (SG) leaks has been partially validated. This was accomplished by comparing code results with data from large leak Sodium/Water Reaction (SWR) tests conducted at the Energy Technology Engineering Center (ETEC). In each of six tests, a Double-Ended Guillotine (DEG) rupture of an SG tube was produced in a sodium-filled SG at typical operating conditions. The primary purpose of these tests was to provide data for validation of the TRANSWRAP (Transient Sodium Water Reaction Analysis Program) code. Primary areas of concern are the leaksite pressure history, pressures at various locations throughout the system, and expulsion of reaction products through the relief system. Agreement with TRANSWRAP results was found to be remarkably good.

1415 (GEFR-SP--171) RESPONSE OF SECONDARY CONTAINMENT TO PRESENCE OF SODIUM AND HYDROGEN. Gleikler, E.L.; Huang, T.C. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Jul 1979. Contract EY-76-C-03-0893. 15p. (CONF-790810--2). Dep. NTIS, PC A02/MF A01.

From 2. international seminar on containment

of fast breeder reactors; Berlin, F.R. Germany (9 Aug 1979).

As part of an effort to demonstrate that the risk to the public from extremely low probability events in liquid metal fast breeder reactors is bound within an acceptable envelope, containment pressurization by sodium and hydrogen was evaluated. Temperature and pressure histories are presented for typical sodium spray and pool fires and sodium vapor reactions. A review of mechanisms for hydrogen generation and recombination as well as limit for flammability and autocatalytic recombination is provided, and general containment design options to reduce risk are discussed.

1416 (GEFR-SP--172) CARBON TRANSPORT IN A BIMETALLIC SODIUM LOOP SIMULATING THE INTERMEDIATE HEAT TRANSPORT SYSTEM OF A LIQUID METAL FAST BREEDER REACTOR. Hampton, L.V.; Spalaris, C.N.; Roy, P. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Apr 1980. Contract EY-76-C-03-0893-031. 20p. (CONF-800401--7). Dep. NTIS, PC A02/MF A01.

From 2. international conference of liquid metal technology for energy systems; Richland, WA, USA (20 Apr 1980).

Carbon transport data from a bimetallic sodium loop simulating the intermediate heat transport system of a Liquid Metal Fast Breeder Reactor are discussed. The results of bulk carbon analyses after 15,000 hours' exposure indicate a pattern of carburization of Type 304 stainless steel foils which is independent of loop sodium temperature. A model based on carbon activity gradients accounting for this behavior is proposed. Data also indicate that carburization of Type 304 stainless steel is a diffusion-controlled process; however, decarburization of the ferritic 2 1/4 Cr-1Mo steel is not. It is proposed that the decarburization of the ferritic steel is controlled by the dissolution of carbides in the steel matrix. The differences in the sodium decarburization behavior of electroslog remelted and vacuum-arc remelted 2 1/4 Cr-1Mo steel are also highlighted.

1417 (GEFR-SP--177) RELIABILITY OF LIQUID METAL BREEDER REACTOR COOLING SYSTEMS. Crellin, G.L.; Bazovsky, I. Sr. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 21 Sep 1979.

Contract EY-76-C-03-0893-032. 29p. (CONF-800103--2). Dep. NTIS, PC A03/MF A01.

From Reliability and maintainability symposium; San Francisco, CA, USA (22 Jan 1980).

The reliability of the cooling systems employed during shutdown of Liquid Metal Fast Breeder Reactor is discussed. A commercial-sized breeder reactor was selected as an example for the reliability modeling of the cooling system. Markov models in the form of state transition diagrams are used to exploit the effect of repairs on system reliability. The actual calculations were performed by a reliability computer program developed by General Electric.

1418 (GEFR-SP--180) OPTIMUM FEW-GROUP CROSS SECTION SET FOR HCDAS IN HETEROGENEOUS LMFBRs. Choong, P.T.; Kujawski, E. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 6 Nov 1979. Contract AM03-76SF00893. 6p. (CONF-800607--32). Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

The loss of sodium in a hypothetical transient with failure to scram proceeds sequentially. In heterogeneous cores, the high power-to-flow fuel regions during the accident will boil first and voiding will propagate to the internal blankets with some time lag. Ideally the multigroup cross sections should take this into account. It would be unrealistic to attempt to generate different multigroup cross sections for each phase of the transient. The objective is to present a simple to implement and yet quite accurate scheme for generating an optimal multigroup cross section set for use in hypothetical core disruptive accidents (HCDA) analysis of heterogeneous LMFBRs. Based on present and the past design procedures, five sets of multigroup cross section sets have been generated for calculating flowing sodium void reactivities of heterogeneous cores.

- 1419 (GEFR-SP--191) CORROSION ISSUES IN STEAM GENERATORS OF THE FAST BREEDER REACTOR SYSTEMS. Spalaris, C.N. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Dec 1979. Contract AT03-76SF7G030. 16p. (CONF-800204--5). Dep. NTIS, PC A02/MF A01.

From Energy-sources technology conference; New Orleans, LA, USA (3 Feb 1980).

Reliability of sodium-heated steam generators is the key element to successful demonstration of Liquid Metal Fast Breeder Reactor. Corrosion resistance of the alloy selected for constructing the steam generators is paramount to high component reliability. A program was undertaken in the US to study the corrosion resistance of 2 1/4Cr-1Mo alloy, the reference construction material for the sodium heated steam generators. Test conditions selected for the experiments represent a wide spectrum of normal and faulted chemistry environments. The results obtained were used to establish (a) the basis for corrosion allowance in design, and (b) an understanding of temperature regimes, metallurgical condition and concentration of impurities that must be avoided in order to achieve high reliability of components. Experimental details, major findings, and significance of these to component reliability are reviewed in this paper.

- 1420 (GEFR-SP--196) PROGRESS OF LMFBR STEAM GENERATOR TESTING IN THE US. Tippets, F.E.; Ford, J.A.; Wright, E.A.; Buschman, H.W. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 1980. Contract FY-76-C-03-0893-010. 24p. (CONF-800401--13). NTIS, PC A02/MF A01.

From 2. international conference of liquid metal technology for energy systems; Richland, WA, USA (20 Apr 1980).

In order to improve assurance of reliable performance, a concerted program of steam generator testing has been underway in the US since the early 1970's, performing both work oriented to the Clinch River Breeder Reactor Plant and work applicable to future plant steam generators. The approach in this program comprises a combination of small-scale and large-scale tests, special feature tests and integral proof tests, and materials tests. This paper describes the overall US program, summarizes some of the main results obtained, and outlines the efforts underway for succeeding phases.

- 1421 (GEFR-SP--198) MEASUREMENT OF IRRADIATION CREEP IN BENDING. McSherry, A.J.; Marshall, J.; Patel, M.R. (General Electric

Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 31 Jan 1980. Contract AT03-76SF71031. 6p. (CONF-800607--25). Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

The major deformation modes in LMFBR fuel channels are bowing caused by neutron flux and temperature gradients and dilation due to stresses imposed by the flowing sodium. In both cases, the stress state of interest is bending. The bulk of irradiation creep data has been generated by simply loaded specimens such as tensile or biaxial pressurized tubes but it is questionable whether this data can be used to predict creep in bending. An irradiation creep experiment using beams loaded in primary bending has been designed to investigate this premise.

- 1422 (GEFR-SP--202) EXPERIMENTAL DETERMINATION OF PLUTONIA AND URANIA SOLUBILITY IN LIQUID METAL. Caputi, R.W.; Adamson, M.G. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). Apr 1980. Contract EY-76-C-03-0893-029. 9p. (CONF-800401--8). Dep. NTIS, PC A02/MF A01.

From 2. international conference of liquid metal technology for energy systems; Richland, WA, USA (20 Apr 1980).

Available data pertaining to the solubilities of uranium and plutonium in liquid sodium at typical reactor coolant temperatures is reviewed. The recurring problem of separating true solubility from micro-spallation/mechanical dispersal effects is discussed and was considered in the experimental design. Descriptions of the filter capsules and analytical techniques that were used are provided. Derived solubility data for both $UO_2(Na_3UO_4)$ and $PuO_2/sub 2-x/(Na_3PuO_4)$ from a powder blended mixed oxide chip (75% UO_2 /25% PuO_2 , initial O:M = 2.00) are given for temperatures in the range 575 to 810°C. Significant levels of both dissolved oxide and micro-particulate were present in the sodium at higher temperatures (700 and 800°C).

- 1423 (GEFR-SP--210) BACKGROUND NOISE MEASUREMENTS ON PROTOTYPE OF SNR 300 STEAM GENERATOR AT HENGLO. Greene, D.A.; Meyer, G.J.A.M.; Ahlgren, F.F.; Van Westerbrugge, J.K. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 1980. Contract EY-76-C-03-0893-031. 10p. (CONF-800401--2). Dep. NTIS, PC A02/MF A01.

From 2. international conference of liquid metal technology for energy systems; Richland, WA, USA (20 Apr 1980).

Acoustic noise amplitudes were measured in a 50 MW helical coil prototype of a steam generator mounted in the Hengelo test facility in a joint technical program between GE/DOE and TNO/NERATOOM. Background noise measurements were completed with test conditions ranging from quiescent (with no sodium or water flow) to high thermal power transfer from the sodium circuit to the water circuit. The acoustic noise pressure on the vessel wall was measured using externally mounted accelerometers. Data were obtained for the effect of sodium flow rate on the acoustic noise amplitude. A preliminary correlation was derived for the acoustic noise generated by nucleate boiling. Leak detection and location were demonstrated.

- 1424 (GEFR-SP--212) ACOUSTIC LEAK DETECTION/LOCATION SYSTEM FOR SODIUM HEATED STEAM GENERATORS. Greene, D.A.; Meneely, D.; Ahlgren, F.F. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems

Dept.). 1980. Contract EY-76-C-03-0893-031. 12p. (CONF-800401--3). Dep. NTIS, PC A02/MF A01.

From 2. international conference of liquid metal technology for energy systems; Richland, WA, USA (20 Apr 1980).

The design of an acoustic leak detection system to monitor for sodium-water reactions in a LMFBR steam generator is described. The detector is based on similar techniques to passive SONAR. It relies on the only absolute parameters associated with a sodium-water reaction: the leak is stationary in space and generates noise. By forming an array from externally mounted sensors and using statistical processing techniques, signals masked by the general background noise can be recovered. Supporting experimental and analytical investigations are also described. Predictions are made of the detection sensitivity of an acoustic leak detection system for an LMFBR.

1425 (GEFR-SP--213) LEAK LOCATION IN A DOUBLE WALL TUBE STEAM GENERATOR. Greene, D.A.; McMurtrie, K.R.; Gaubatz, D.C. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 1980. Contract EY-76-C-03-0893-031. 8p. (CONF-800401--4). Dep. NTIS, PC A02/MF A01.

From 2. international conference of liquid metal technology for energy systems; Richland, WA, USA (20 Apr 1980).

An experimental and analytical study was made of an acoustic location system for use with double wall tube steam generators. A conceptual design of an acoustic leak location system was developed. This system has the potential for detecting the third fluid gas escaping from the annulus between the double walled tubes into the sodium. The acoustic system uses an array of accelerometers mounted onto the outside of the steam generator vessel. Electronic circuitry selects a number of accelerometers, forms them into an array which focuses within the vessel. The volume of the vessel is sequentially scanned for anomalous acoustic energy.

1426 (GEFR-SP--00183) INFLUENCE OF THE MAIN CONDENSER ON SHRS RELIABILITY. Elerath, J.G. (General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.). 21 Dec 1979. Contract EY-76-C-03-0893-032. 6p. (CONF-800607--4). Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

Upon shutdown of a Large Breeder Reactor, decay and latent heat must be removed to keep core temperatures below specified design limits. Usually this function will be performed using the plant's main condenser as the heat sink. If it is not available, however, heat can be removed by an auxiliary Shutdown Heat Removal System (SHRS). In support of the DOE/ RRT Safe Shutdown Reliability Program, quantitative reliability studies were performed to investigate the relationships among various auxiliary SHRS configurations and the main condenser. The results of these studies show that the relative reliabilities achieved are a strong function of the auxiliary configurations selected; also that credit for the main condenser has significant reliability benefits in certain cases.

1427 (GRS-F--63) REPORT ON THE RESEARCH PROJECTS IN THE FIELD OF REACTOR SAFETY SPONSORED BY THE FEDERAL MINISTRY FOR RESEARCH AND TECHNOLOGY. PERIOD UNDER REPORT: JANUARY 1 - MARCH 31, 1978. (Gesellschaft fuer

Reaktorsicherheit m.b.H. (GRS), Koeln (Germany, F.R.)). Sep 1978. 520p. (In German). Dep. NTIS (US Sales Only), PC A23/MF A01.

Investigations on the safety of Light Water Reactors (LWR) being performed in the framework of the Research Program Reactor Safety (RS - Projects) are sponsored by the BMFT (Federal Minister for Research and Technology), Bundesminister fuer Forschung und Technologie. Objective of this program is to investigate in greater detail the safety margins of nuclear power plants and their systems and the further development of safety technology. Besides the investigations of LWR tasks first projects on the safety of FBR type reactors are sponsored by the BMFT. The GRS (Reactor Safety Association), Gesellschaft fuer Reaktorsicherheit mbH, by order of the BMFT, informs continuously of the status of such investigations by means of quarterly and annually publication of progress reports within the series GRS - F - Fortschrittsberichte (GRS - F - Progress Reports). Each progress report represents a compilation of individual reports about objectives, the work performed, the results, the next steps of the work etc. The individual reports are prepared in a standard form by the contractors themselves as a documentation of their progress in work. The individual reports are arranged according to the amended LWR Safety Research Program of the BMFT. Another table contents uses the same classification system as applied in the Nuclear Safety Index of the CEC Communities and the OECD. (orig./HP) 891 HP.

1428 (GRS-F--66) REPORTS ON RESEARCH PROJECTS IN THE FIELD OF REACTOR SAFETY SPONSORED BY THE FEDERAL MINISTRY FOR RESEARCH AND TECHNOLOGY. PERIOD UNDER REPORT: APRIL 1 TO JUNE 30, 1978. (Gesellschaft fuer Reaktorsicherheit m.b.H. (GRS), Koeln (Germany, F.R.)). Sep 1978. 673p. (In German). Dep. NTIS (US Sales Only), PC A09/MF A01.

Investigations on the safety of Light Water Reactors (LWR) being performed in the framework of the Research Program Reactor Safety (RS - Projects) are sponsored by the BMFT (Federal Minister for Research and Technology), der Bundesminister fuer Forschung und Technologie. Objective of this program is to investigate in greater detail the safety margins of nuclear power plants and their systems and the further development of safety technology. Besides the investigations of LWR tasks first projects on the safety of FBR type reactors are sponsored by the BMFT. The GRS (Reactor Safety Association), Gesellschaft fuer Reaktorsicherheit mbH, by order of the BMFT, informs continuously of the status of such investigations by means of quarterly and annually publication of progress reports within the series GRS - F Fortschrittsberichte (GRS - F - Progress Reports). Each progress report represents a compilation of individual reports about objectives, the work performed, the results, the next steps of the work etc. The individual reports are prepared in a standard form by the contractors themselves as a documentation of their progress in work. The individual reports are arranged according to the amended LWR Safety Research Program of the BMFT. Another table contents uses the same classification system as applied in the Nuclear Safety Index of the CEC Communities and the OECD. (orig./HP) 891 HP.

1429 (GRS-F--70) REPORT ON THE PROJECTS IN THE FIELD OF REACTOR SAFETY SPONSORED BY THE FEDERAL MINISTRY FOR RESEARCH AND TECHNOLOGY. PERIOD UNDER REPORT: 1ST JULY TO 30TH SEPTEMBER 1978. (Gesellschaft fuer Reaktorsicherheit

m.b.H. (GRS), Koeln (Germany, F.R.)). Dec 1978. 183p. (In German). Dep. NTIS (US Sales Only), PC A24/MF A01.

Investigations on the safety of Light Water Reactors (LWR) being performed in the framework of the Research Program Reactor Safety (RS - Projects) are sponsored by the BMFT (Federal Minister for Research and Technology), Bundesminister fuer Forschung und Technologie. Objective of this program is to investigate in greater detail the safety margins of nuclear power-plants and their systems and the further development of safety technology. Besides the investigations of LWR tasks first projects on the safety of advanced reactors are sponsored by the BMFT. The GRS (Reactor Safety Association), Gesellschaft fuer Reaktorsicherheit mbH, by order of the BMFT, informs continuously of the status of such investigations by means of quarterly and annually publication of progress reports within the series GRS - F - Fortschrittsberichte (GRS - F - Progress Reports). Each progress report represents a compilation of individual reports about objectives, the work performed, the results, the next steps of the work etc. The individual reports are prepared in a standard form by the contractors themselves as a documentation of their progress in work and published by the FB (Research Coordination Department), Forschungsbetreuung at the GRS, within the framework of general information of the progress in reactor safety research. The individual reports are arranged according to the amended LWR Safety Research Program of the BMFT which will appear in the near future. Another table contents uses the same classification system as applied in the Nuclear Safety Index of the CEC and the OECD.

- 1430 (GRS-F--74) REPORTS ON RESEARCH PROJECTS SPONSORED BY THE FEDERAL MINISTER FOR RESEARCH AND TECHNOLOGY IN THE FIELD OF REACTOR SAFETY. PROGRESS REPORT FOR THE PERIOD FROM OCTOBER 1 TO DECEMBER 31, 1978. (Gesellschaft fuer Reaktorsicherheit m.b.H. (GRS), Koeln (Germany, F.R.)). Mar 1979. vp. (In German). NTIS (US Sales Only).

Investigations on the safety of Light Water Reactors (LWR) being performed in the framework of the Research Program Reactor Safety (RS - Projects) are sponsored by the BMFT (Federal Minister for Research and Technology), Bundesminister fuer Forschung und Technologie. Objective of this program is to investigate in greater detail the safety margins of nuclear power plants and their systems and the further development of safety technology. Besides the investigations of LWR tasks first projects on the safety of FBR type reactors are sponsored by the BMFT. The GRS (Reactor Safety Association), Gesellschaft fuer Reaktorsicherheit mbH, by order of the BMFT, informs continuously of the status of such investigations by means of quarterly and annually publication of progress reports within the series GRS - F - Fortschrittsberichte (GRS - F - Progress Reports). Each progress report represents a compilation of individual reports about objectives, the work performed, the results, the next steps of the work etc. The individual reports are prepared in a standard form by the contractors themselves as a documentation of their progress in work. The individual reports are arranged according to the amended LWR Safety Research Program of the BMFT. Another table contents uses the same classification system as applied in the Nuclear Safety Index of the CEC European Communities and the OECD.

- 1431 (GRS-F--79) REPORTS ON RESEARCH

PROJECTS SPONSORED BY THE FEDERAL MINISTER FOR RESEARCH AND TECHNOLOGY IN THE FIELD OF REACTOR SAFETY. PERIOD UNDER REPORT 1ST JANUARY TO 31ST MARCH. PROGRESS REPORT. (Gesellschaft fuer Reaktorsicherheit m.b.H. (GRS), Koeln (Germany, F.R.)). Jun 1979. vp. (In German). Fachinformationszentrum Karlsruhe.

Investigations on the safety of Light Water Reactors (LWR) being performed in the framework of the Research Program Reactor Safety (RS - Projects) are sponsored by the BMFT (Federal Minister for Research and Technology), Bundesminister fuer Forschung und Technologie. Objective of this program is to investigate in greater detail the safety margins of nuclear power plants and their systems and the further development of safety technology. Besides the investigations of LWR tasks first projects on the safety of FBR type reactors are sponsored by the BMFT. The GRS (Reactor Safety Association), Gesellschaft fuer Reaktorsicherheit mbH, by order of the BMFT, informs continuously of the status of such investigations by means of quarterly and annually publication of progress reports within the series GRS-F-Fortschrittsberichte (GRS-F-Progress Reports). Each progress report represents a compilation of individual reports about objectives, the work performed, the results, the next steps of the work etc. The individual reports are prepared in a standard form by the contractors themselves as a documentation of their progress in work. The individual reports are arranged according to the amended LWR Safety Research Program of the BMFT. Another table contents uses the same classification system as applied in the Nuclear Safety Index of the CEC Communities and the OECD.

- 1432 (HEDL--77-106) SUMMARY REPORT OF FFTF PROTOTYPE PRIMARY PUMP SODIUM TEST PROGRAM. Buonamici, R. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1977. Contract EY-76-C-14-2170. 84p. AT.

A summary of the high temperature in-sodium testing and test results of the full-size Fast Flux Test Facility (FFTF) prototype primary pump in the Sodium Pump Test Facility (SPTF) at the Liquid Metal Engineering Center (LMEC) is presented. The overall objectives of the test program, which spanned the period from July 1975 to April 1977, were to demonstrate the highest confidence in and the integrity of the design of the pump prior to its actual operation in the FFTF. Design modifications made to the pump following the hydrostatic bearing seizure during the initial thermal transient test program were successfully validated by extensive testing.

- 1433 (HEDL--6232) PROGRAM OUTLINE UNDER-SODIUM VIEWING. Holtink, N.C. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1977. Contract EY-76-C-14-2170. 8p. AT.

An outline is presented of a program with the following objectives: (1) provide technical management and required basic development for completing the design, fabrication, and Composite Reactor Component Test Activity (CRCTA) testing of a prototype Under-Sodium Viewing (USV) system and complete all necessary modifications to the prototype system and make available for FFTF use; (2) identify the under-sodium viewing and ranging requirements for the Clinch River Breeder Reactor (CRBR) and develop the special ultrasonic concepts, techniques, and equipment needed to satisfy these requirements; and (3) develop transducers, electronics, and mechanisms to improve reactor inservice inspection capabilities. Specific

improvements include image enhancement, in-vessel storage of future USV systems and real time imaging. Advanced systems will provide capability for inservice inspection and surveillance.

- 1434 (HEDL--6297) US--UK TECHNICAL EXCHANGE BREEDER REACTOR FUELS: FUEL PIN CHEMISTRY. Woodley, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 14 Oct 1977. Contract EY-76-C-14-2170. vp. AT.

Included are thermochemical data for use in analysis of mixed-oxide fuel chemical behavior and data on in-core behavior of materials to control fuel-cladding chemical interaction. (DG)

- 1435 (HEDL--6313) SODIUM IMPURITY MONITORING. Barton, G.B.; McCown, J.J.; Keough, R.F. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-07-1375. 5p. AT.

Two basic impurity monitoring methods are in use in the U.S. Breeder Reactor Program. The first involves instrumentation which performs measurements in-situ with real time data output from on-line sensors. The second involves obtaining some type of sample which is removed from the system and analyzed in a chemistry lab with the data obtained representing the impurity level as it was one or more days previously. These methods are reviewed and planned research is described.

- 1436 (HEDL--6314) MASS AND ACTIVITY TRANSPORT. Anantatmula, R.P.; Brehm, W.F.; Chulos, L.E.; McGuire, J.C. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-07-1375. 7p. AT.

LMFBR-related work accomplished in two programs, the Radioactivity Control Technology (RCT) Program, and the Advanced Alloy Sodium Compatibility Program, is described. The objective of the RCT Program is to develop and demonstrate methods for reducing the transport of radioactive species to maintenance zones in primary heat transport systems. Both corrosion and fission products are considered. The Compatibility Program is designed to test and qualify cladding and duct alloys.

- 1437 (HEDL--6557) NONDESTRUCTIVE TESTING METHODS AND EQUIPMENT: SECTION 3.0. SEMIANNUAL REPORT TO OAK RIDGE NATIONAL LABORATORY, OCTOBER 1978--MARCH 1979. ADVANCE COPY. Michaels, T.E.; Horn, J.E.; Boehmer, W.D.; Trantow, R.L. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1979. Contract EY-76-C-14-2170. 15p. Dep. NTIS, PC A02/MF A01.

A system configuration has been determined for scan control, data acquisition, and real-time analysis for ultrasonic inspection of a weld in the secondary loop at FFTF. This system will significantly decrease time requirements for inspection and reduce data storage by utilizing a host computer to coordinate activities of dedicated stations. Acceptance testing of the Ultrasonic Scanning Mechanism (USM) was successfully completed. Results indicate that the USM is a well designed and well built mechanism. All motions met or exceeded specifications and operated smoothly without mechanical binding or interferences. Elevated temperature tests of USM components were completed. Stepping motors, encoders, limit switches, and cooling fans were shown to be capable of intermittent operation over

extended time periods at temperatures as high as 93°C (200°F). An improved transducer configuration has been determined for faster and more accurate ultrasonic scanning. Individual elements are being characterized in the laboratory for this multi-element transducer. Transducer work at Argonne National Laboratory is planned to develop a wedge suitable for the FFTF inspection.

- 1438 (HEDL--6583) EXPERIMENTAL IRRADIATION OF FRACTURE MECHANICS SPECIMENS IN FAST REACTORS. (Westinghouse Hanford Co., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 22p. Dep. NTIS, PC A02/MF A01.

Two LMFBR irradiation facilities are, or will be, providing the great majority of irradiation space in support of fracture mechanics characterization efforts: the Experimental Breeder Reactor-II (EBR-II), and the Fast Flux Test Facility (FFTF). Current plans call for the EBR-II, which has been operating for a number of years, to be directed to other testing programs sometime after the startup of FFTF, presently scheduled for late 1979. However, since there may be a period of time during which both facilities are operating and available for specimen irradiation, features and characteristics of both will be discussed herein, although the emphasis will be on the FFTF. The purpose of this report is to provide the necessary background for considering possible collaborative programs on fracture toughness and crack growth of irradiated materials between USDOE and UKAEA, to be conducted under appropriate exchange agreements.

- 1439 (HEDL--6617) COMPUTERIZED INTERFERENCE ELIMINATION REPORT: FFTF CELLS 489, 422E, 529A, AND 529B. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 258p. Dep. NTIS, PC A12/MF A01.

Design information is presented concerning the possible piping interferences in FFTF cells 489, 422E, 529A, and 529B.

- 1440 (HEDL--6629) FFTF PREPRODUCTION MECHANICAL SNUBBER QUALIFICATION TEST PROGRAM FOR WESTINGHOUSE/HANFORD. (Hanford Engineering Development Lab., Richland, WA (USA)). 10 Nov 1978. Contract EY-76-C-14-2170. 384p. AT.

Performance testing data are presented for FFTF reactor seismic restraints.

- 1441 (HEDL--SA-1595-FP) BOILUP THRESHOLD FOR THE BOTTLED-UP TRANSITION PHASE POOL. Martin, F.J. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1978. Contract EY-76-C-14-2170. 7p. (CONF-781105--95). Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

Since the inception of the hypothesized transition phase, for the late stages of a postulated LMFBR accident, there has been a continual effort to characterize the anticipated conditions of such a hypothetical state. To date, several techniques and methods have been employed to analyze the potential for energetic criticality. As part of this effort, an arbitrary criterion of monotonic dispersiveness has been employed as the measure of diminished recriticality potential. The various attempts to demonstrate monotonic dispersiveness have included experimental demonstrations, theoretical approaches, and integrated analysis using both. As part of this

treatment, flow regime maps have been devised as a convenient method for inferring the state of dispersiveness. They included bubbly, churn turbulent, foam and drop fluidized regimes. Of these, foam and drop fluidized regimes were considered the most dispersive. The main thrust of the analysis to date, including flow regime maps, relates primarily to the open pool configuration. However, the bottled configuration may be the pertinent geometry. To date, no reliable escape path has been demonstrated for the advanced stages of core disruption, although strong potential escape mechanisms have been identified and are currently being analyzed. The bottled pool is examined in this paper.

- 1442 (HEDL-BR--5853-C-4(Rev.1)) FFTF REPORT: DRILLED-IN EXPANSION BOLTS UNDER STATIC AND ALTERNATING LOAD. Tsang, H.S. (Bechtel Power Corp., San Francisco, CA (USA)). 1 Oct 1976. Contract EY-76-C-14-2170. 92p. Dep. NTIS, PC A05/MF A01.

The objective of this investigation was to establish the allowable design loads (tension, shear, and combined load) for expansion bolts to be installed in regular 4000 psi mix concrete, Magnetite 5000 psi mix concrete, and Steel Shot 5000 psi mix concrete at the Fast Flux Test Facility. The test loads included Static Loads and Alternating Loads which simulate the dynamic loads of vibratory equipment and dynamic earthquake loads.

- 1443 (HEDL-S/A--1552-FP) FISSION GAS BEHAVIOR IN MIXED-OXIDE FUEL DURING OVERPOWER AND THERMAL TRANSIENT TESTS. Randklev, E.H.; Hinman, C.A. (Hanford Engineering Development Lab., Richland, WA (USA)). 1 Nov 1978. Contract EY-76-C-14-2170. 25p. (CONF-790306--24). Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

Results are presented from two complementary Hanford Engineering Development Laboratory (HEDL) programs which are investigating fission gas behavior in irradiated mixed-oxide fuel. The combined purpose of these programs is to define steady-state and transient fission gas behavior at axial fuel pin positions as a function temperature, radial and microstructural position and mechanisms of transient gas release. The following discussion, unless otherwise noted, applies to fuel irradiated at powers from approx. 20 kW/m to approx. 40 kW/m and for burnups of approx. 40 to approx. 60 MWd/kg.

- 1444 (HEDL-SA--135-3) FFTF DUMP HEAT EXCHANGER DESIGN AND DEVELOPMENT. Seber, E.C.; Mancuso, R.A.; Jansen, W.D.; Mahaffey, M.K. (Hanford Engineering Development Lab., Richland, WA (USA)). [nd]. Contract EY-76-C-14-2170. 55p. AT.

A brief summary of the history of procurement of the Dump Heat Exchangers for FFTF is presented. It outlines the various organizations and respective roles. A description of the design basis and features of the large main heat transport system dump heat exchangers is included. Design evolution is discussed and testing performed as part of the design process, including finned tube qualification, header hydraulic tests, air-pressure drop tests, insulated panel tests and friction/wear testing, is summarized. Manufacturing experience, including development of an integral finned tube, development of welding procedures for joining the finned tubes, and header fabrication is also

presented. Testing results obtained with the prototype unit are described, as are additional feature model tests that were performed to define possible methods of improving heat transfer performance for the plant units. A summary of construction experience, status of plant modifications and a description of expected plant operation including decay heat removal is included.

- 1445 (HEDL-SA--464) LINEAR HEAT RATING FOR INCIPIENT FUEL MELTING IN UO_2 -- PuO_2 FUEL. Leggett, R.D.; Ballard, E.O.; Baker, R.B.; Horn, G.R.; Dutt, D.S. (Hanford Engineering Development Lab., Richland, WA (USA)). 1974. Contract EY-76-C-14-2170. 6p. (CONF-741116--2). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; Washington, DC, USA (Nov 1974).

Results are presented for an experiment (HEDL-P-19) conducted in EBR-II to determine the effect of the initial, cold, diametral fuel-cladding gap size (3.2 to 10.0 mils) on the linear heat rating to incipient fuel melting, $Q'/\text{sub m}$, in mixed oxide (75% UO_2 --25% PuO_2) fuel pins under very rapid LMFBR startup conditions. Eight 0.250 in. OD X 0.016 in. wall and eight 0.230 in. OD X 0.015 in. wall, He bonded pins clad with type 316 stainless steel (20% CW) were irradiated in an encapsulated pin subassembly in Row 2 of the EBR-II. Each fuel pellet was measured to ± 0.2 mil, weighed and its position in the 13 1/2 in. long fuel column recorded. Most of the pellets were centerless ground to a predetermined size. Four pins--two of each size--contained pellets that were sintered to size. The fuel density and O/M of the 0.230 in. OD pins were 90.75% TD and 1.96, respectively. The 0.250 in. OD pins contained fuel with a density of 92.40% and an O/M of 1.97. This test was conducted to verify that no fuel melting will occur in the FTR at 115% of the 400 MW rating.

- 1446 (HEDL-SA--737) INFLUENCE OF BURNUP ON HEAT-RATING-TO-MELTING FOR UO_2 -- PuO_2 FUEL. Leggett, R.D.; Baker, R.B.; Dutt, D.S.; Chastain, S.A. (Westinghouse Hanford Co., Richland, WA (USA)). 1974. Contract EY-76-C-14-2170. 7p. (CONF-741116--1). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; Washington, DC, USA (Nov 1974).

This paper summarizes the results of an irradiation test (HEDL-P-20) conducted to evaluate the influence of burnup on the linear heat rating to incipient fuel melting, $Q'/\text{sub m}$, for 75% UO_2 --25% PuO_2 fuel. The very short term irradiation was conducted in Row 1 of the EBR-II to simulate 115% of FTR full power using an irradiation history similar to that used for the HEDL-P-19 test. Prototypic FTR fuel pins (0.230 in. O.D., type 316, 20% cw stainless steel) were included that were preirradiated to 0% BU, 0.3% BU, 0.7% BU, and 1% BU at a simulated 100% FTR power. One 0.250 in. O.D. pin (PNL-2-5) clad with type 304 stainless steel was also included that was preirradiated to approx. 7% BU. Fuel-cladding diametral gap sizes were varied from 3.5 to 7.5 mils (the FTR range) by centerless grinding the pellets to different diameters, helium and 80He + 20 Xe fill gases were used, and as-built fuel microstructure was also a test variable. The preirradiation of the 0.230 in. O.D. pins was done in Row 3 of the EBR-II for one reactor cycle and one-half reactor cycle, respectively. All seven of the 0.230 in. O.D., 'fresh', pins had extensive fuel melting, but fuel melting was indicated in only one of the preirradiated pins. This pin was fabricated with a 3.5 mil gap and preirradiated to approx.

0.3% BU. Similar pins preirradiated to 0.7% BU and 1% BU showed no melting. The high burnup PNL-2-5 pin (0.250 in. O.D.) also had fuel melting, but this is believed to be due to the 20% higher heat generation rate in this larger pin, rather than to a reduction in thermal performance at very high burnup.

- 1447 (HEDL-SA--838) DOCUMENT CONTROL AND INFORMATION RETRIEVAL SYSTEM FOR THE FAST FLUX TEST FACILITY (FFTF). Theo, M.G. (Westinghouse Hanford Co., Richland, WA (USA)). 1 Mar 1976. Contract EY-76-C-14-2170. 26p. (CONF-7606165--1). Dep. NTIS, PC A03/MF A01. From Special Libraries Association; Denver, CO, USA (7 Jun 1976).

A description is given of the FFTF Document Control and Information Retrieval System. The system utilizes a mini-computer along with various microfilm equipment and is designed to accommodate an anticipated 50 million pages of text and 750,000 drawings. The system is simple, uncluttered, eliminates duplication, and provides quick retrievability of documents for all technical and administrative personnel.

- 1448 (HEDL-SA--859(FP)) IN-REACTOR CORROSION BEHAVIOR OF STAINLESS STEEL CLADDING IN HIGH TEMPERATURE SODIUM. Weber, J.W. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1976. Contract EY-76-C-14-2170. 9p. (CONF-760503--20). Dep. NTIS, PC A02/MF A01.

From International conference on liquid metal technology in energy production; Champion, PA, USA (3 May 1976).

Sodium-cladding chemical interaction has been studied on fuel pins clad with 20% cold worked type 316 stainless steel and irradiated in the EBR-II at temperatures up to 705°C and for exposures to 5300 hours. The measured corrosion rate of the cladding surface immediately above the top of the fuel column was 12.5 μm per year at 690°C. The loss of Ni at 700°C resulted in the formation of a ferrite layer approximately 5 μm thick. A zone depleted in Ni and Cr extends into the austenite from the ferrite-austenite interface an additional distance of approximately 15 μm . No large changes in volumetric average carbon or nitrogen were observed.

- 1449 (HEDL-SA--897) BUNDLE-DUCT CLEARANCE OF HEDL MIXED OXIDE SUBASSEMBLIES. Hill, R.G.; Jackson, R.J. (Hanford Engineering Development Lab., Richland, WA (USA)). 1976. Contract EY-76-C-14-2170. 17p. (CONF-761103--31). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; Washington, DC, USA (15 Nov 1976).

One parameter of interest bearing upon LMFBR fuel pin performance is the "clearance" between the fuel pin bundle and its associated duct. This clearance is determined by calculating the distance between a theoretically tight bundle and the inner surface of the duct wall. Because the individual fuel pins have varying amounts of bending and bowing, the pin bundle is a springy array which expands to fill the duct. Therefore, the calculated clearance exists as a somewhat randomly distributed porosity throughout the pin bundle. A computer code is described which calculates the amount of clearance which is distributed. This computer code, POROSITY, considers the mechanisms of thermal expansion, neutron induced swelling, and creep. These mechanisms each cause a dimensional change in the fuel pin bundle and duct.

- 1450 (HEDL-SA--1082) FFTF CORE AND PRIMARY SODIUM CIRCUIT INSTRUMENTATION. Day, C.K. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1975. Contract EY-76-C-14-2170. 41p. (CONF-760137--3). Dep. NTIS, MF A01.

From International Atomic Energy Agency working party; Risley, UK (27 Jan 1976).

Portions of document are illegible.

This paper presents plans, engineering parameters, and some test results for several FFTF core and primary sodium circuit instrument systems. Systems to be discussed include temperature, flow, pressure, leak detectors, level sensors, fuel failure monitoring, sodium impurity analysis and cover gas monitors. Results of recent demonstration tests of the FFTF Under-Sodium Viewing and Ranging system are also presented.

- 1451 (HEDL-SA--1154) GAS TAGGING FOR FUEL FAILURE LOCATION USING XENON AND KRYPTON. Goheen, M.W.; Plucinski, C.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 14p. (CONF-770687--1). Dep. NTIS, PC A02/MF A01.

From The American Society of Mass Spectrometry; Washington, DC, USA (2 Jun 1977).

The FFTF (Fast Flux Test Facility) nearing completion at the Hanford Engineering Development Laboratory near Richland, Washington, under the direction of The Westinghouse Hanford Co., is designed to test components and materials for the development of Liquid Metal Fast Breeder Reactors. During the initial stages of operation, the FFTF will be shut down when a fuel or control pin has a detectable cladding leak. To achieve maximum reactor operating time, monitoring methods for detecting and locating leaking pins were developed. Monitoring techniques to signal a leak are based on changes of specific activity levels of individual xenon and krypton fission product isotopes. Gas tagging is the method used to locate the failed pin. Isotopically differing tags of a gas or gases can be used as a fingerprint technique for sealed assemblies such as fuel or control pins with mass spectrometer detection to locate a specific leaker. 104 unique tags were blended for the 80 fuel assemblies and 24 control assemblies.

- 1452 (HEDL-SA--1164FP) TRITIUM RELEASE FROM FAST NEUTRON IRRADIATED BORON CARBIDE. Hollenberg, G.W. (Westinghouse Hanford Co., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 17p. (CONF-770416--6). Dep. NTIS, PC A03/MF A01.

From 79. annual meeting of American Ceramic Society; Chicago, IL, USA (23 Apr 1977).

A high-energy neutron reaction with boron produces tritium. In the LMFBR control material, B₄C, most of the tritium that is generated remains in the pellets. Potential retention mechanisms are discussed. 5 figures.

- 1453 (HEDL-SA--1167) STRESS-AFFECTED MICROSTRUCTURAL DEVELOPMENT AND CREEP-SWELLING INTERRELATIONSHIP. Brager, H.R.; Garner, F.A.; Gilbert, E.R.; Flinn, J.E.; Wolfer, W.G. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1977. Contract EY-76-C-14-2170. 48p. (CONF-770641--12). Dep. NTIS, PC A03/MF A01.

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Macroscopic measurement of the deformations

arising from swelling and creep during neutron irradiation indicate that both processes are dependent on the magnitude and possibly the sign of the applied stress state. Current modeling efforts also indicate that a strong interaction exists between swelling and creep through the stress state. Because the macroscopic distortions arise from the integrated microscopic strains associated with specific microstructural elements, the effect of applied stress on microstructural development has been studied.

- 1454 (HEDL-SA--1171-FP) MICROSTRUCTURE CHARACTERIZATION OF ADVANCED OXIDE FUEL. Rasmussen, D.E.; Gerber, E.W.; McCord, R.B. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 65p. (CONF-770416--7). Dep. NTIS, PC A04/MF A01.

From 79. annual meeting of American Ceramic Society; Chicago, IL, USA (23 Apr 1977).

Preirradiation porosity, grain size, and microcomposition characteristics are presented for selected advanced oxide ($\text{PuO}_2\text{-UO}_2$) LMFBR developmental fuels fabricated for irradiation testing in EBR-II. Quantitative microscopy, electron microprobe analysis, and a recently developed quantitative autoradiographic technique are utilized to relate microstructure characteristics to fabrication parameters.

- 1455 (HEDL-SA--1175) U. S. PROGRAMS ON REFERENCE AND ADVANCED CLADDING/DUCT MATERIALS. Bennett, J.W.; Holmes, J.J.; Laidler, J.J. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1977. Contract EY-76-C-14-2170. 21p. (CONF-770641--13). Dep. NTIS, PC A02/MF A01.

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Two coordinated national programs are presently in place in the United States for development of reference and advanced cladding and duct alloys for near-term and long-term LMFBR applications. A number of government, industrial and university laboratories are active participants in these two ERDA-sponsored programs. The programs are administered by ERDA through a task group organization, with each task group representing a particular technical activity and the membership of the task group drawn from among the laboratories with active involvement in that activity. Technical coordination of the two programs is provided by the Hanford Engineering Development Laboratory. The National Reference Cladding and Duct Program is charged with the responsibility for development of the required technology to permit full utilization of the reference material, 20 percent cold-worked Type 316 stainless steel, in early LMFBR core applications. The current schedule calls for full evaluation of FFTF-related design base data prior to full-power operation of FFTF in early 1980, followed by a confirmation in early 1983 of reference material performance capabilities for initial-core CRBRP applications. Comprehensive evaluation of reference material performance to commercial plant goal fluence levels will be complete by 1985. The National Advanced Alloy Development Program was instituted in 1974 with the objective to develop, by 1986, advanced cladding and duct materials compatible with advanced fuel systems having peak burnup capabilities up to 150 MWD/kg and doubling times of 15 years or less. Screening of a large number of potential alloys was completed in mid-1975, and there are presently 16 candidate alloys under active investigation.

- 1456 (HEDL-SA--1177-FP) ABSORBER MATERIALS FOR FAST REACTOR CONTROL APPLICATIONS. Birney, K.R.; Pitner, A.L.; Bourquin, R.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1977. Contract EY-76-C-14-2170. 20p. (CONF-770611--29). Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; New York, NY, USA (12 Jun 1977).

This paper provides a brief description of design requirements for LMFBR absorber materials, and addresses some of the more important material property and performance considerations. The relative merits and disadvantages of the various candidates are compared with respect to these parameters, and the present status of the LMFBR absorber development program is addressed.

- 1457 (HEDL-SA--1189-FP) CORRELATION FOR BORON CARBIDE HELIUM RELEASE IN FAST REACTORS. Basmajian, J.A.; Pitner, A.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1977. Contract EY-76-C-14-2170. 14p. (CONF-770622--7). Dep. NTIS, PC A02/MF A01.

From Air Pollution Control Association meeting; Toronto, Canada (19 Jun 1977).

An empirical helium correlation for the helium release from boron carbide has been developed. The correlation provides a good fit to the experimental data in the temperature range from 800 to 1350°K, and burnup levels up to 80×10^{20} captures/cm³. The correlation has the capability of extrapolation to 2200°K (3500°F) and 200×10^{20} captures/cm³. In this range the helium release rate will not exceed the generation rate.

- 1458 (HEDL-SA--1190FP) HYDROGEN JET RECOMBINATION UNDER POSTULATED LMFBR ACCIDENT CONDITIONS. Wierman, R.W. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 11p. (CONF-770611--27). Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; New York, NY, USA (12 Jun 1977).

Certain conditions may be postulated in LMFBR risk assessments for which the potential of hydrogen release to the reactor containment building needs to be evaluated. The inherent self-ignition characteristics of hydrogen jets entering the air atmosphere of the reactor containment building should be understood for such analyses. If hydrogen jets were to self-ignite (recombine) at the source where they enter the reactor containment building, then undesirable hydrogen accumulation would not occur. Therefore, experiments have been conducted investigating the phenomena associated with the recombination of hydrogen jets under conditions similar to those postulated for LMFBR studies. The data presented define the conditions required for self-ignition of the hydrogen jets.

- 1459 (HEDL-SA--1191(Rev.)) TRIBOLOGY: RESEARCH AND DEVELOPMENT NEEDS IN ADVANCED ENERGY TECHNOLOGY. Johnson, R.N. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 30p. (CONF-770237--1). Dep. NTIS, PC A03/MF A01.

From Workshop on tribology in energy technology; Washington, DC, USA (7 Feb 1977).

Poorly controlled wear and friction affects energy conservation, material conservation, and the reliability and safety of mechanical

systems, and is estimated to cost U.S. industries \$16 billion/yr. ERDA's National Friction, Wear, and Self-Welding Program and its accomplishments are described. This program includes studies of wear and friction problems in high temperature and unusual environments, e.g., as experienced by LMFBR components, and common to much technology involving energy conversion using fossil-fuel, geothermal, nuclear, and solar resources. Program activities for tribology information handling and wear and friction testing are discussed. (LCL)

- 1460 (HEDL-SA--1205) CORE DESIGN METHODS FOR ADVANCED LMFBRs. Chandler, J.C.; Marr, D.R.; McCurry, D.C.; Cantley, D.A. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1977. Contract EY-76-C-14-2170. 17p. (CONF-770611--28). Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; New York, NY, USA (12 Jun 1977).

The multidiscipline approach to advanced LMFBR core design requires an iterative design procedure to obtain a closely-coupled design. HEDL's philosophy requires that the designs should be coupled to the extent that the design limiting fuel pin, the design limiting duct and the core reactivity lifetime should all be equal and should equal the fuel residence time. The design procedure consists of an iterative loop involving three stages of the design sequence. Stage 1 consists of general mechanical design and reactor physics scoping calculations to arrive at an initial core layout. Stage 2 consists of detailed reactor physics calculations for the core configuration arrived at in Stage 1. Based upon the detailed reactor physics results, a decision is made either to alter the design (Stage 1) or go to Stage 3. Stage 3 consists of core orificing and detailed component mechanical design calculations. At this point, an assessment is made regarding design adequacy. If the design is inadequate the entire procedure is repeated until the design is acceptable.

- 1461 (HEDL-SA--1216(Rev.)) FAST FLUX TEST FACILITY STARTUP PLAN. Carlisle, C.S.; Noordhoff, B.H. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 37p. (CONF-770822--7). Dep. NTIS, PC A03/MF A01.

From American Nuclear Society meeting; Chattanooga, TN, USA (8 Aug 1977).

The principal features of the FFTF startup plan are described and the startup schedule is discussed. The present status of startup testing is summarized.

- 1462 (HEDL-SA--1220) EFFECT OF SUPPORT CLAMPS ON ELEVATED TEMPERATURE PIPING. Hyde, L.L. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 28p. (CONF-770807--58). Dep. NTIS, PC A03/MF A01.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Two finite-element models were used to analyze the effects of local support clamp loads in the FFTF piping systems. The results indicate that clamp influences on pipe stresses and strains may be significant.

- 1463 (HEDL-SA--1221) SEMI-AUTOMATED URANIUM ANALYSIS BY A MODIFIED DAVIES--GRAY PROCEDURE. Swanson, G.C. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977.

Contract EY-76-C-14-2170. 16p. (CONF-7706109--1). Dep. NTIS, PC A02/MF A01.

From Northwest regional meeting; Portland, OR, USA (15 Jun 1977).

To rapidly and reliably determine uranium in fuel materials a semi-automated implementation of the Davies-Gray uranium titration was developed. The Davies-Gray method is essentially a three step procedure. First uranium is reduced quantitatively from +6 valence to +4 valence by excess of iron (II) in strong phosphoric acid in the absence of nitrite. Prior to the uranium reduction nitrite is destroyed by addition of sulfamic acid. In the second step iron (II) is selectively oxidized to iron (III) by nitric acid in the presence of Mo (VI) catalyst. Finally after dilution to reduce phosphate concentration, the uranium is titrated to U (VI) by standard dichromate. The original sluggish colorimetric endpoint determination used by Davies and Gray is seldom used since New Brunswick Laboratory discovered that addition of vanadium (IV) just prior to titration sufficiently improves reaction rate to allow a potentiometric endpoint determination. One of the advantages of the Davies-Gray uranium titration is that it is quite specific for uranium, most common impurity elements do not interfere with the analysis, and specifically high levels of Pu, Th, and Fe are tolerated.

- 1464 (HEDL-SA--1225FP(Rev.)) MICROSTRUCTURAL CHARACTERIZATION OF ADVANCED FERROUS ALLOYS EXPOSED TO LIQUID SODIUM. Anantatmula, R.P.; Mayhan, S.J.; White, P.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1977. Contract EY-76-C-14-2170. 10p. (CONF-770739--1). Dep. NTIS, PC A02/MF A01.

From 10. annual technical meeting of the International Metallographic Society; Houston, TX, USA (19 Jul 1977).

The property requirements for breeder reactor fuel cladding materials are extremely stringent. Alloys with low sodium corrosion rates are required to avoid loss of strength from wall thinning at reactor operating temperatures. Data obtained on the following alloys are described: A-286, Inconel 706, Nimonic PE16, AISI 316, M-813 (sintered), Inconel 718, AISI 310, and Alloy 330. Tests were carried out at 700°C for 2000 hours at a sodium velocity of 6 m/s and an oxygen content in sodium of 1 ppm. Corrosion rates and major elements depleted are tabulated. It is concluded that A-286, Inconel 706, and Nimonic PE16 are highly promising commercial alloys for cladding purposes on the basis of corrosion resistance. 5 figures, 1 table. (RWR)

- 1465 (HEDL-SA--1230-S) FFTF GAS PROCESSING SYSTEMS. Halverson, T.G. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 21p. (CONF-770512--22). Dep. NTIS, PC A02/MF A01.

From Symposium on management of low level radioactive waste; Atlanta, GA, USA (23 May 1977).

The design and operation of the two radioactive gas processing systems at the Fast Flux Test Facility (FFTF) exemplifies the concept that will be used in the first generation of Liquid Metal Fast Breeder Reactors (LMFBR's). The two systems, the Radioactive Argon Processing System (RAPS) and the Cell Atmosphere Processing System (CAPS), process the argon and nitrogen used in the FFTF for cover gas on liquid metal systems and as inert atmospheres in steel lined cells housing sodium equipment. The RAPS specifically

processes the argon cover gas from the reactor coolant system, providing for decontamination and eventual reuse. The CAPS processes radioactive gases from inerted cells and other liquid metal cover gas systems, providing for decontamination and ultimate discharge to the atmosphere. The cryogenic processing of waste gas by both systems is described.

- 1466 (HEDL-SA--1241) LMFBR ACCIDENT CONSEQUENCES ASSESSMENT. Simpson, D.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 39p. (CONF-770625--6). Dep. NTIS, PC A03/MF A01.

From International conference on nuclear systems reliability engineering and risk assessment; Gatlinburg, TN, USA (20 Jun 1977).

Deterministic analyses of hypothetical LMFBR accidents have been applied in safety evaluations of FFTF and CRBRP. These analyses involve the application of complex inter-related computational techniques to assess a wide range of circumstances more extreme than any which are realistically foreseen. Such analyses are well suited to be employed in combination with probabilistic techniques for risk assessment.

- 1467 (HEDL-SA--1268) IN-REACTOR BEHAVIOR OF MATERIALS USED TO CONTROL FUEL-CLADDING CHEMICAL INTERACTION. Wilson, C.N.; Gibby, R.L.; Lawrence, L.A.; Weber, E.T. (Hanford Engineering Development Lab., Richland, WA (USA)). 22 Aug 1977. Contract EY-76-C-14-2170. 27p. (CONF-770958--1). Dep. NTIS, PC A03/MF A01.

From American Ceramic Society fall meeting; Hyannis, MA, USA (26 Sep 1977).

Niobium, vanadium, chromium and titanium metals were irradiation tested in various configurations as candidate oxygen buffer/getter materials in LMFBR fuel pins. Test results indicate in-situ oxidation of buffer/getter materials with specific materials and configurations preventing cladding attack. Some placement configurations were ineffective due to low oxygen mobility. A relative preference for buffer/getter reaction with fission product cesium was also noted.

- 1468 (HEDL-SA--1286) CLADDING DEVELOPMENT AND CHARACTERIZATION IN THE UNITED STATES. Lobsinger, R.J. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 27p. (CONF-770694--1). Dep. NTIS, PC A03/MF A01.

From U.S. Japan exchange meeting; Sunnyvale, CA, USA (27 Jun 1977).

During the past decade, an extensive program of cladding development has been carried out at HEDL in support of the FFTF/Breeder Reactor Program. This activity has resulted in the establishment of specification, fabrication, and characterization requirements for breeder reactor fuel cladding. In establishing these requirements, input has been derived from vendor development programs, evaluation studies, and reviews with other national laboratories. The major fabrication requirements of the current specification as derived from the development program and the testing required to assure product conformance and characterize cladding behavior are described.

- 1469 (HEDL-SA--1288) SAFETY ASPECTS OF FAST REACTOR FUELS. Hanson, J.E. (Hanford Engineering Development Lab., Richland, WA

(USA)). 1977. Contract EY-76-C-14-2170. 85p. Dep. NTIS, PC A05/MF A01.

Design requirements and performance characteristics of fuel pins for LMFBR reactors under accident conditions are discussed.

- 1470 (HEDL-SA--1292-S) EFFECTS OF STOICHIOMETRY ON CLADDING ATTACK IN MIXED-OXIDE FUELS TO approx. = 3.6 AT% BURNUP. Hata, D.C.; Lawrence, L.A.; Weber, J.W. (Hanford Engineering Development Lab., Richland, WA (USA)). 26 May 1977. Contract EY-76-C-14-2170. 14p. (CONF-771109--99). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

Effects of initial stoichiometry on the character and extent of fuel-cladding chemical interaction (FCCI) were established for mixed-oxide fuels irradiated to peak burnups of approximately 3.6 at.%. The present LMFBR design basis considers an initial reduction of 50 microns (2 mils) in cladding thickness to compensate for possible FCCI. This thickness loss represents a significant part of the total cladding wastage allowance for the LMFBR reference fuels design. The reduction or elimination of FCCI would allow a corresponding reduction in cladding thickness resulting in an improvement in the breeding ratio and a reduction in system doubling time.

- 1471 (HEDL-SA--1294S) ANALYSIS OF FUEL-CLADDING CHEMICAL INTERACTION AT HIGH BURNUP. Lawrence, L.A.; Weber, J.W.; Devary, J.L. (Hanford Engineering Development Lab., Richland, WA (USA)). 26 May 1977. Contract EY-76-C-14-2170. 6p. (CONF-771109--101). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

The HEDL fuel-cladding chemical interaction (FCCI) wastage correlation has been revised to incorporate additional data from the HEDL P-23 high temperature test series at peak fuel pin burnups of approximately 90.0 MWd/kgM. The previous HEDL correlation was developed from a data base of fuel pins with peak burnups of approximately 50.0 MWd/kgM. Therefore, in order to predict the amount of cladding thickness loss to be expected at current anticipated fuel pin lifetimes in the range of 90 to 100 MWd/kgM a major extrapolation of the actual irradiation experience was necessary. To provide the needed data for the extension of the correlation to high burnups the irradiation of the HEDL P-23A subassembly in EBR-II was extended to a peak fuel pin burnup of approximately 90.0 MWd/kgM. An advanced matrix type reaction was observed in the high burnup fuel pin. The character and extent of the interaction made it possible to estimate the location of the original cladding inner surface to determine cladding thickness loss due to FCCI by measuring the depth of penetration. Therefore, it was necessary to measure total cladding thickness for the high burnup pin. In order to place the entire data set on the same basis, cladding thickness was also measured for the lower burnup pins. Cladding thickness was measured on 225X photographs of the polished cladding at 16 equally spaced locations on the circumference of each sample. From these measurements an average cladding thickness was calculated.

- 1472 (HEDL-SA--1298) FFTF SCALE-MODEL CHARACTERIZATION OF FLOW-INDUCED VIBRATIONAL RESPONSE OF REACTOR INTERNALS. Ryan, J.A.; Julyk, L.J. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 98p. (CONF-770933--2).

Dep. NTIS, PC A05/MF A01.

From International Atomic Energy Agency, International working group on fast reactors; Argonne, IL, USA (20 Sep 1977).

As an integral part of the Fast Test Reactor Vibration Program for Reactor Internals, the flow-induced vibrational characteristics of scaled Fast Test Reactor core internal and peripheral components were assessed under scaled and simulated prototype flow conditions in the Hydraulic Core Mockup. The Hydraulic Core Mockup, a 0.285 geometric scale model, was designed to model the vibrational and hydraulic characteristics of the Fast Test Reactor. Model component vibrational characteristics were measured and determined over a range of 36 percent to 111 percent of the scaled prototype design flow. Selected model and prototype components were shaker tested to establish modal characteristics. The dynamic response of the Hydraulic Core Mockup components exhibited no anomalous flow-rate dependent or modal characteristics, and prototype response predictions were adjudged acceptable.

1473 (HEDL-SA--1299-S) CLADDING BREACHES IN MIXED OXIDE FUEL PINS. Weber, J.W.; Almasy, M.Y. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1977. Contract EY-76-C-14-2170. 6p. (CONF-771109--104).

Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

Breaches of the cladding have occurred on eleven fuel pins from the HEDL mixed oxide (75 w/o UO_2 - 25 w/o PuO_2) irradiation tests in EBR-II. These pins were in 19 or 37 pin subassemblies being irradiated as part of the HEDL Steady State and Run-to-Cladding Breach activities for the FFTF Reference Fuel Program. The linear powers and temperatures of these tests were within the FFTF Steady State design envelopes. Cladding materials included solution-annealed type 304 and 20 percent or 30 percent cold-worked type 316 stainless steel. All pins were fabricated to the reference specifications current at the time for the FFTF driver fuel, except for those with the 30 percent cold-worked cladding. The objectives of the studies and examinations of the breached pins are to: (1) establish the burnup capability for reference fuel, (2) identify cladding breach mechanisms, (3) determine the effects of cladding breach on neighboring fuel pins, (4) evaluate the effects of short term postbreach residence in reactor on fuel-coolant interaction, and (5) provide data for failure criteria. Examination and characterization have been completed on nine of the eleven breached pins, with two pins currently being examined. In addition, 46 neighbors to the breached pins have been evaluated for possible effects as a consequence of the breach.

1474 (HEDL-SA--1300FP) INTERSUBASSEMBLY INCOHERENCIES AND GROUPING TECHNIQUES IN LMFBR HYPOTHETICAL OVERPOWER ACCIDENT. Wilburn, N.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1977. Contract EY-76-C-14-2170. 39p. (CONF-771109--97).

Dep. NTIS, PC A03/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

A detailed analysis was made of the FTR core using the 100-channel MELT-IIIA code. Results were studied for the transient overpower accident (where 0.5\$/sec and 1\$/sec ramps) and in which the Damage Parameter and the Failure Potential criteria were used. Using the information obtained from these series of runs, a new method of grouping the subassemblies into channels has been developed. Also, it was

demonstrated that a 7-channel representation of the FTR core using this method does an adequate job of representing the behavior during a hypothetical disruptive transient overpower core accident. It has been shown that this new 7-channel grouping method does a better job than an earlier 20-channel grouping. It has also been demonstrated that the incoherency effects between subassemblies as shown during the 76-channel representation of the reactor can be adequately modeled by 7-channels, provided the 7-channels are selected according to the criteria stated in the report. The overall results of power and net reactivity were shown to be only slightly different in the two cases of the 7-channel and the 76-channel runs. Therefore, it can be concluded that any intersubassembly incoherencies can be modeled adequately by a small number of channels, provided the subassemblies making up these channels are selected according to the criteria stated.

1475 (HEDL-SA--1306S) EFFECTS OF LOCAL CESIUM CONCENTRATIONS ON MIXED-OXIDE FUEL BEHAVIOR. Karnesky, R.A. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 6p. (CONF-771109--106). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

Local peaks in cesium concentrations were found to influence the thermal-mechanical behavior of mixed-oxide fuel pins at high burnup. A prototypic fuel pin was irradiated in EBR-II at a peak linear power of approximately 400 W/cm (12 kW/ft) with peak cladding temperatures of approximately 540°C (1000°F) to a peak burnup of approximately 8.7 at. %. The mixed-oxide pellet fuel (75 wt % UO_2 , 25% PuO_2) had an initial oxygen-to-metal ratio of 1.956 and a planar smeared density of 90% of theoretical. The pin was clad with .041 cm (0.016 in.) thick, .635 cm (0.250 in.) diameter, solution-annealed type 316 stainless steel. In the region of the upper UO_2 insulator the high cesium concentration and attendant cesium-uranium reaction zone are found to be restricted to a narrow region on the circumference of the first insulator pellet. This reaction zone resulted in extensive fragmentation of the insulator pellet approximately .24 cm (.1 in.) above the fuel-insulator interface. Densification of the insulator pellet fragment adjacent to the fuel column appears to have occurred and may be due to increased temperatures resulting from the increased thermal resistance of the crack. The cesium-uranium reaction region appears to consist of two zones, a high density area where the uranium insulator pellet has undergone grain boundary attack and dissolution and a lower density zone with a more porous cesium-uranium reaction product.

1476 (HEDL-SA--1316) UNCERTAINTIES IN FAST BREEDER REACTOR FUEL PIN CLADDING STRESS AND STRAIN CALCULATIONS. Sutherland, W.H. (Hanford Engineering Development Lab., Richland, WA (USA)). [nd]. Contract EY-76-C-14-2170. 27p. (CONF-780609--18). Dep. NTIS, PC A03/MF A01.

From ASME/CSME pressure vessels and piping conference; Montreal, Canada (25 Jun 1978).

Two LMFBR material response phenomena, irradiation creep and swelling, make the prediction of cladding stress and strain uncertain. Temperature gradients through the cladding wall result in high predicted swelling gradients because of the extreme sensitivity of swelling to temperature. If the

phenomenological material model for irradiation swelling does not assume a stress effect, then large stresses are calculated due to the large swelling gradients. Detailed analyses have been performed on selected fuel pins where the loading on the cladding has been restricted to fission gas pressure, temperature gradients, and swelling gradients. Various combinations of phenomenological material models for irradiation creep and swelling were used. Large variations in calculated cladding stresses and strains result from the above combination of material models. Resolution of the correct material model for irradiation creep and swelling could be achieved if the residual stress distribution and swelling distribution through the cladding thickness were known. Experimental investigations are underway to develop a procedure for measuring the residual stress and swelling distribution through the cladding wall.

- 1477 (HEDL-SA--1321) DESIGN AND ECONOMICS OF LARGE LMFBRs. Omberg, R.P.; Cantley, D.A.; Last, G.A.; Squire, A.; Lowery, V.W. (Hanford Engineering Development Lab., Richland, WA (USA); Energy Research and Development Administration, Washington, DC (USA). Div. of Reactor Development and Demonstration). Jun 1977. Contract EY-76-C-142-2170. 31p. (CONF-771123--3). Dep. NTIS, PC A03/MF A01. From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The design of an LMFBR core is a complex process which is strongly dependent upon design philosophy, design criteria, material properties, and the calculational procedure involved. The development of a core design with optimum characteristics, whether it be a minimum doubling time, a minimum power cost, or a minimum sodium void worth, requires a close coupling of the disciplines of reactor physics, thermohydraulics, mechanical design, and fuel pin performance. This is true regardless of whether the fuel is oxide, carbide, nitride, or metal, and regardless of whether the fuel cycle is Pu/U or U-233/Th.

- 1478 (HEDL-SA--1324-F) GROUND TEMPERATURES SURROUNDING A MOLTEN FUEL POOL. Peak, R.D. (Hanford Engineering Development Lab., Richland, WA (USA)). 22 Jun 1977. Contract EY-76-C-14-2170. 3p. (CONF-771109--96). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

In the analysis of the consequences of a hypothetical meltdown accident in an LMFBR, it is important to estimate the final location of the molten fuel pool in the concrete and ground underlying the reactor vessel. The GROWS program and the AYER program have been developed to calculate the final location of the molten fuel pool as the culmination of the transient analysis of this unusual Stefan problem but these programs require extensive computational resources. The solution is provided to the concrete and ground temperatures surrounding the stationary fuel pool and the related heat flux from the pool to the ground surface outside the containment building. This solution can be used to estimate the final location of the fuel pool and to check the end results of the sophisticated programs.

- 1479 (HEDL-SA--1325) DESIGN CONFIGURATIONS FOR CONTROLLING SODIUM VOID WORTH IN LARGE FBR CORES. Chandler, J.C.; Marr, D.R. (Hanford Engineering Development Lab., Richland, WA

(USA)). Sep 1977. Contract EY-76-C-14-2170. 25p. (CONF-771109--105). Dep. NTIS (US Sales Only), PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

Analyses of sodium voiding in small fast reactors with plutonium-uranium fuel--such as FFTF--indicate that a relatively small positive reactivity effect will result. Similar analyses for large fast breeder reactors (FBRs) in the 1200 MWe range, however, indicate that a much larger positive reactivity effect will occur. It is important, therefore, to develop design configurations and methods of defining those configurations so as to minimize the sodium void effect without adversely affecting the breeding and economic performance. A set of design configurations with these characteristics is presented in the paper. Since no practical experience is available for defining an acceptable value for sodium void worth, the voiding criterion was chosen to be whole-core sodium void worth roughly equivalent to that of CRBR, which is on the order of three dollars. This value was chosen on the assumption that CRBR could be licensed with such a void worth. Four reactor systems were evaluated for sodium void worth. The systems represented a direct scale up of the FFTF subassembly to 1200 MWe, a homogeneous core utilizing the FTR reference fuel pin in an optimized subassembly, a heterogeneous core utilizing the same fuel pin, and an alternate fueled design based on the thermal characteristics of the CRBR assembly. The designs are compared in terms of breeding and economic performance and sodium void worth.

- 1480 (HEDL-SA--1326) TRITIUM PERMEATION THROUGH OXIDIZED 304 AND 316 STAINLESS STEEL. McGuire, J.C. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1978. Contract EY-76-C-14-2170. 27p. (CONF-780213--10). Dep. NTIS, PC A03/MF A01. From 107. AIME meeting; Denver, CO, USA (26 Feb 1978).

This work was completed during the course of a cooperative program on Tritium Transport in Liquid Metal Cooled Reactors. This work considers conditions encountered in operating reactors. It involves hydrogen, deuterium and tritium permeation measurements on "clean" and oxidized surfaces. The tritium measurements were restricted to permeation rates from sodium and through the test membrane at concentrations typical of EBR-II reactor experience. The hydrogen and deuterium work measured permeation rates at gas pressures from 13-1300 Pa (0.1 to 10 Torr). Permeation rates were measured for tritium permeation through oxidized 304 and 316 stainless steel. The tritium was injected into hot flowing sodium at concentrations near present sodium cooled reactor operating levels. Permeation data are presented as a function of temperature, oxidation time and tritium activity level in the sodium. Permeation rates were determined at 590, 540 and 480°C and at tritium concentrations varying from 1 to 40 x 10⁻⁹ CiT₂/gmNa (3.4 x 10⁻⁸ to 1 x 10⁻⁶ ppm). At 540°C tritium diffusion rates for both oxidized steels were in the range 1-2 x 10⁻⁴ cc-mm/hr-cm²-atm^{1/2}. Response at this concentration varied as the square root of pressure. However hydrogen (H₂) and deuterium permeation tests at 1300 to 13 Pa (10 to 0.1 Torr.) showed linear response to changes in this higher pressure range.

- 1481 (HEDL-SA--1327-FP) PERFORMANCE OF THORIUM FUELED FAST BREEDERS. Marr, D.R.; Cantley, D.A.; Chandler, J.C.; McCurry, D.C.; Omberg, R.P. (Hanford Engineering Development

Lab., Richland, WA (USA)). Oct 1977. Contract EY-76-C-14-2170. 38p. (CONF-771109--103). Dep. NTIS, PC A03/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

Three studies were performed to evaluate the breeding ratio of fast breeders containing thorium. In a study of a small breeder, thorium metal and thorium oxide core designs were found to have similar breeding ratios. The slight advantage exhibited by the metal design was not considered significant since the design was based on a limited amount of thorium metal swelling data. In a study of a 1200 MWe plant, a plutonium-uranium oxide design was compared to a uranium-thorium design. The uranium-thorium design had a lower breeder ratio, but also had a negative sodium void effect. In the third study, the effect of replacing thorium oxide radial blankets with thorium metal radial blankets was evaluated. This was found to have little effect on the breeding ratio.

1482 (HEDL-SA--1327-S) PERFORMANCE OF THORIUM FUELED FAST BREEDERS. Marr, D.R.; Cantley, D.A.; Chandler, J.C.; McCurry, D.C.; Haffner, D.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1977. Contract EY-76-C-14-2170. 7p. (CONF-771109--102). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

Three studies were made of the performance characteristics of thorium-containing reactors. The first of these compared two small breeder designs, one using thorium oxide, the other using thorium metal, both with ^{233}U enrichment. The second compared a plutonium-uranium oxide 1200 MWe design to a uranium-thorium 1200 MWe design. The third compared the performances of thorium metal and thorium oxide radial blankets for three large breeders using uranium or thorium oxide cores and axial blankets. Results are presented and discussed.

1483 (HEDL-SA--1329) REACTOR PHYSICS PARAMETERS OF ALTERNATE FUELED FBR CORE DESIGNS. Haffner, D.R.; Hardie, R.W. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 31p. (CONF-771109--112). Dep. NTIS, PC A03/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

Nuclear non-proliferation considerations have resulted in renewed interest in the thorium fuel cycle. Reactor physics parameters of a typical 1200-MW(e) Fast Breeder Reactor (FBR) design were compared when U-233 is substituted for Pu as a fissile fuel and when Th-232 is substituted for U-238 as a fertile fuel. When the U-238 in the blanket is replaced with Th-232, the reactor physics parameters are relatively unchanged. However, replacing U-238 in the core with Th-232 increases the critical mass by 11 to 15% and decreases the breeding ratio by 0.13 to 0.16. In addition, replacing the Pu in the core with U-233 decreases the critical mass by 4 to 6% and decreases the breeding ratio by 0.13 to 0.16. Both of the changes in the core make the sodium void coefficient more negative.

1484 (HEDL-SA--1333) TESTING OF FFTF FUEL HANDLING EQUIPMENT. Coleman, D.W.; Grazzini, E.D.; Hill, L.F. (Energy Research and Development Administration, Washington, DC (USA); Westinghouse Hanford Co., Richland, WA (USA)). Jul 1977. Contract EY-76-C-14-2170. 43p. (CONF-771120--24). Dep. NTIS, PC A03/MF A01.

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

The Fast Flux Test Facility has several manual/computer controlled fuel handling machines which are exposed to severe environments during plant operation but still must operate reliably when called upon for reactor refueling. The test programs for two such machines--the Closed Loop Ex-Vessel Machine and the In-Vessel Handling Machine--are described. The discussion centers on those areas where design corrections or equipment repairs substantiated the benefits of a test program prior to plant operation.

1485 (HEDL-SA--1343) PULSED MAGNETIC WELDING OF BREEDER REACTOR FUEL PIN END CLOSURES. Brown, W.F.; Bendas, J.; Olson, N.T. (Hanford Engineering Development Lab., Richland, WA (USA); Maxwell Labs., Inc., San Diego, CA (USA)). [nd]. Contract EY-76-C-14-2170. 14p. (CONF-780406--4). Dep. NTIS, PC A02/MF A01.

From 59. AWS annual meeting; New Orleans, LA, USA (3 Apr 1978).

High-frequency, high-intensity pulsed magnetic fields were used in an experimental program to produce leak proof fuel pin end closures on stainless steel and nickel base alloys. Metallographic examination of these solid state welds between thin walled fuel pin cladding tubes and end plugs showed a wavy or rippled appearance which is characteristic of high velocity impact bonding. Leak tests and mechanical tests showed that the end closure welds were leak proof and stronger than the cladding tube material.

1486 (HEDL-SA--1348) DEVELOPMENT OF ULTRASONIC EXAMINATION METHODS FOR AUSTENITIC STAINLESS STEEL WELD INSPECTION. Mech, S.J.; Michaels, T.E.; Emmons, J.S.; Sugiyama, M.M. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1977. Contract EY-76-C-14-2170. 29p. Dep. NTIS, PC A03/MF A01.

Signal acquisition and analysis techniques were developed to permit locating notches placed in an austenitic stainless steel pipe weld. Ultrasonic techniques presented here utilized refracted longitudinal and shear inspection beams with a dual element, pitch-catch transducer. Sources of grain noise and the metallurgical implications of inspecting austenitic stainless steel welds are reviewed. Data were analyzed in both time and frequency domains, and a simple frequency moment algorithm was developed which is sensitive to defect signal and somewhat insensitive to grain noise. Data were also used to train an adaptive learning network (ALN) and the results obtained demonstrate only slight sensitivity to grain noise.

1487 (HEDL-SA--1350) NONDESTRUCTIVE ANALYSIS OF IRRADIATED FUELS. Dudey, N.D.; Frick, D.C. (Exxon Nuclear Co., Inc., Richland, WA (USA); Westinghouse Hanford Co., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 20p. (CONF-771036--14). Dep. NTIS, PC A02/MF A01.

From 2. ASTM-EURATOM symposium on reactor dosimetry: dosimetry methods for fuels, cladding and structural materials; Palo Alto, CA, USA (2 Oct 1977).

The principal nondestructive examination techniques presently used to assess the physical integrity of reactor fuels and cladding materials include gamma-scanning, profilometry, eddy current, visual inspection,

rod-to-rod spacing, and neutron radiography. LWR fuels are generally examined during annual refueling outages, and are conducted underwater in the spent fuel pool. FBR fuels are primarily examined in hot cells after fuel discharge. Although the NDE techniques are identical, LWR fuel examinations emphasize tests to demonstrate adherence to technical specification and reliable fuel performance; whereas, FBR fuel examinations emphasize aspects more related to the relative performance of different types of fuel and cladding materials subjected to variable irradiation conditions.

- 1488 (HEDL-SA--1351) FFTF REACTOR CHARACTERIZATION PROGRAM. Daughtry, J.W.; Bennett, R.A.; Bunch, W.L.; McElroy, W.N.; King, T.L. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 16p. (CONF-771036--10). Dep. NTIS, PC A02/MF A01.

From 2. ASTM-EURATOM symposium on reactor dosimetry: dosimetry methods for fuels, cladding and structural materials; Palo Alto, CA, USA (2 Oct 1977).

Preparations are under way for the initial startup and testing of the Fast Flux Test Facility (FFTF). The FFTF Reactor Characterization Program is that part of the startup test plan that deals with the determination of the neutron, gamma ray and thermal hydraulic characteristics of the reactor. This program encompasses measurements and calculations of: neutron spectra, flux and fluence; gamma-ray spectra, dose and heating; fission rate distributions; capture rate distributions; other reaction rates of interest; fission product yields; and thermal hydraulic data. Measurements of these parameters will be made in the reactor core and reflectors, will extend vertically downward to the vicinity of the core support structure and upward to the top of the sodium pool, and will extend radially outward to include in-vessel fuel storage locations and the cavity between the reactor vessel and the concrete wall.

- 1489 (HEDL-SA--1354-FP) INELASTIC ANALYSIS OF PIPELINE IN FFTF HEAT TRANSPORT SYSTEM. Chen, W.L.; Weiner, E.O. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 27p. (CONF-780609--15). Dep. NTIS, PC A03/MF A01.

From ASME/CSME pressure vessels and piping conference; Montreal, Canada (25 Jun 1978).

The detailed inelastic (elastic-plastic-creep) analysis is presented for the Primary Crossover Pipe Line in the Fast Flux Test Facility (FFTF), subjected to the combined deadweight and thermal loads. It demonstrates the inelastic structural responses of the complete pipe run and satisfaction of ASME Code requirements at elevated temperatures. The analytical tool used for this analysis is the general purpose nonlinear finite element program, MARCCDC. The pipe line is modeled using the combinations of closed section beam elements and 3-node elbow elements. Results of creep relaxation under multiaxial loading of the piping systems are compared with those of uniaxial loading in a simple two-bar model. Creep-fatigue damage evaluations are calculated by following the procedures of ASME Code Case 1331-5.

- 1490 (HEDL-SA--13638) US--UK TECHNICAL EXCHANGE BREEDER REACTOR FUELS. FUEL PIN CHEMISTRY (SUPPORTING INFORMATION). (Hanford Engineering Development Lab., Richland, WA (USA)). 14 Oct 1977. Contract EY-76-C-14-

2170. 55p. (CONF-7710139--1). Dep. NTIS, PC A04/MF A01.

From US-UK technical exchange; Sunnyvale, CA, USA (14 Oct 1977).

The following information covering the two topic areas of fuel pin chemistry; fuel-cladding chemical interaction and cesium behavior will not be formally exchanged with the UK representatives prior to the October 14, 1977, meeting at GE in Sunnyvale, California. This information will only be presented to the UK representatives at the exchange meeting if they offer new information of comparable technical value. Data are presented on irradiation behavior and performance of mixed oxide LMFBR fuels. (DG)

- 1491 (HEDL-SA--1368) DESIGN EVOLUTION OF FFTF HEAT TRANSPORT SYSTEM PUMPS. Young, M.W.; Nixon, D.R.; Absher, K.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1977. Contract EY-76-C-14-2170. 61p. AT.

The Fast Flux Test Facility (FFTF) Heat Transport System (HTS) pumps have been completed and are installed. The basic designs for the prototype, primary, and secondary pumps are described in this report. The descriptions include changes based on design reviews, high temperature and thermal transient considerations, sub-component development testing, and pump testing. Emphasis has been placed on changes resulting from pump tests in water and sodium test loops. Prototype and plant unit configurations are compared to show applicability of sodium test data to the plant units. The pump bearing failure experienced during thermal transient testing is also described, along with the changes required to prevent recurrence. Special prototype pump modifications were made to obtain data verifying adequacy of the design changes to the hydrostatic bearing. These changes are significant to future development programs, as are potential design improvements intended to reduce sensitivity to long-term dimensional instability and severe thermal transients.

- 1492 (HEDL-SA--1369) FFTF FUEL HANDLING SYSTEM, PORT CLOSURES AND SEALS: FEATURES AND EXPERIENCE. Hornberger, D.W. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 41p. AT.

A brief description of the FFTF Fuel Handling System is presented, with emphasis on the ex-reactor to in-reactor vessel handling machines and techniques. The description includes the Closed Loop Ex-Vessel Machine, In-Vessel Handling Machine, Floor Valve, In-Vessel Storage Module, and Fuel Transfer Port and its Floor Valve Adapter. Each item of equipment is reviewed with regard to design features which are important to the functional operation both when acting alone and in combination with the interfacing component. Where applicable, the different requirements imposed on the component during reactor operation and refueling are identified, including the expected response to off-normal events. In addition, the results of some of the test work which has been done on the fuel handling equipment relative to the operational requirements are described and results of purge techniques, measurements of inflatable seal leakages, an evaluation of long term diffusion of sodium vapor into annular spaces, and identification of potential design improvements in the machine operations are included.

- 1493 (HEDL-SA--1370) FFTF INSTRUMENT TREE

DESIGN AND DEVELOPMENT. Packard, R.B. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1977. Contract EY-76-C-14-2170. 31p. AT.

A general description of the instrument tree is presented along with a discussion of its function in the FFTF. This includes the relationship of the instrument tree to the overall operation of the reactor. This is followed by a description of the design criteria used to develop the unit. The areas covered include flow and temperature monitoring of the core, secondary holddown of the core components, support for the control rod system, radiation, thermal and major structural requirements. Following each area is a discussion of the design features used to satisfy the requirements. A discussion of the instrumentation used to monitor the sodium flow and temperature of the core is presented. The instrumentation required to operate and monitor the instrument tree itself is covered. Fabrication of the prototype and installed units is covered with emphasis on areas such as the welding of the in-reactor portion which is essentially an 18,000 pound (8172 Kg), 32 foot (9.75 meters) long solid stainless steel weldment. The extensive testing performed on the prototype unit in air and sodium and the design changes required for the plant units are covered.

1494 (HEDL-SA--1371) FFTF EM SERVICE PUMP DESIGN, APPLICATION, AND TESTING. Jacobson, G.; Carter, E.H.; Hallinan, G.J.; Eichelberger, R.L. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EX-76-C-14-2170. 25p. (CONF-771217--10). Dep. NTIS, PC A02/MF A01.

From Seminar on LMFBR components; Canoga Park, CA, USA (5 Dec 1977).

Portions of this document are illegible.

During 1975, HEDL selected Atomic International to review the design of three classes of EM Pumps (L-30, L-100, L-200) and to fabricate a total of fifteen pumps. One of the L-100 pumps was shipped to the liquid Metal Engineering Center (LMEC) for sodium testing in the small components test loop (SCTL). The paper presents the HEDL design criteria, discusses the resulting design, and reviews the test results against the predictions.

1495 (HEDL-SA--1374) STANDARDIZATION OF DOSIMETRY AND DAMAGE ANALYSIS WORK FOR U.S. LWR, FBR, AND MFR DEVELOPMENT PROGRAM. McElroy, W.N.; Doran, D.G.; Gold, R.; Morgan, W.C.; Grundl, J.A.; McGarry, E.D.; Kam, F.B.K.; Swank, J.H.; Odette, G.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1978. Contract EY-76-C-14-2170. 46p. (CONF-771036--11). Dep. NTIS, PC A03/MF A01.

From 2. ASTM-EURATOM symposium on reactor dosimetry: dosimetry methods for fuels, cladding and structural materials; Palo Alto, CA, USA (2 Oct 1977).

The accuracy requirements for various measured/calculated exposure and correlation parameters associated with current dosimetry and damage analysis procedures and practices depend on the accuracy needs of reactor development efforts in testing, design, safety, operations, and surveillance programs. Present state-of-the-art accuracies are estimated to be in the range of ± 2 to 30 percent (1 sigma), depending on the particular parameter. There now appears to be international agreement, at least for the long term, that most reactor fuels and materials programs will not be able to accept an uncertainty greater than about ± 5 percent (1 sigma). The current status of

dosimetry and damage analysis standardization work within the U.S. for LWR, FBR and MFR is reviewed in this paper.

1496 (HEDL-SA--1375) EBR-II VALIDATED, KEY FISSION PRODUCT YIELDS FOR FAST REACTOR APPLICATION. Kellogg, L.S.; Davis, A.I.; Lippincott, E.P.; Ruggles, J.M. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 15p. (CONF-781002--1). Dep. NTIS (US Sales Only), PC A02/MF A01.

Relative fission yields were measured for three different locations in the row 4 "Test Region" of the EBR-II reactor. Correlation of the relative fission yields to the measured average energy (anti E) and the measured ^{137}Cs $^{238}\text{U}/^{235}\text{U}$ spectral indices have been made. The measured relative fission yields for selected fission products from ^{235}U , ^{238}U , ^{239}Pu and ^{237}Np have been compared with those values reported by the Interlaboratory Reaction Rate (ILRR) program, EBR-II fast reactor yields from destructive analysis and summation, and the March 1977 version of ENDF/B-V.

1497 (HEDL-SA--1380) EXPERIMENTAL AND THEORETICAL STUDIES ON WATER AND GAS RELEASE FROM HEATED CONCRETE. McCormack, J.D.; Postma, A.K. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 11p. (CONF-771108--8). Dep. NTIS, PC A02/MF A01.

From Post accident heat removal information exchange meeting; Chicago, IL, USA (2 Nov 1977).

Procedures currently used in licensing of nuclear facilities require analysis of postulated accidents which are more severe than "design basis" events. For breeder program reactors, some accidents in this severe category involve spillage of sodium coolant onto concrete protected by a steel liner. Heat transfer through the liner heats the concrete, causing a part of the mixing water and other gases to be driven off. These gases would add to pressure in the containment atmosphere, and the water vapor can form hydrogen if it contacts sodium. Evaluations of containment integrity for such postulated accidents will be aided by knowledge of how much water and other gases are released from heated concrete. The report presents results of a research effort at Hanford Engineering Development Laboratory designed to improve understanding of the release of water and gases from heated concrete.

1498 (HEDL-SA--1381) PRELIMINARY RESULTS OF CSTF AEROSOL BEHAVIOR TEST, AB-1. Hilliard, R.K.; McCormack, J.D.; Hassberger, J.A.; Muhlestein, L.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1977. Contract EY-76-C-14-2170. 36p. (CONF-7710102--1). Dep. NTIS, PC A03/MF A01.

From Aerosol transport and release review meeting; Silver Springs, MD, USA (27 Oct 1977).

A large-scale aerosol behavior test (AB-1) was performed in the Containment Systems Test Facility (CSTF) containment vessel using sodium oxide aerosol generated by a pool of sodium burning in air. The purpose was to characterize the aerosol properties and compare the experimental results with computer code predictions. The 20-meter high CSTF vessel is by far the largest ever used in aerosol agglomeration studies and is approximately half-scale of large commercial reactor containment buildings for the parameters affecting particle agglomeration and settling.

- 1499 (HEDL-SA--1382-FP) THERMALLY INDUCED STRESSES AND FRACTURES IN BORON CARBIDE PELLETS. Hollenberg, G.W. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 23p. (CONF-780545--5). Dep. NTIS (US Sales Only), PC A02/MF A01.

From Meeting of the American Ceramic Society; Detroit, MI, USA (6 May 1978).

Neutron absorber materials are used for the control of power levels in fast breeder reactors. In these applications, boron carbide has emerged as the best overall absorber material. Absorber pins in the Fast Test Reactor (FTR) consist of stacks of hot-pressed boron carbide pellets encapsulated in stainless steel tubes. The tubes are arranged in a triangular array inside a hexagonal duct. Experience has dictated the use of stoichiometric boron carbide with a nominal density of 92% of theoretical. Due to the internal heating which results from neutron captures in the pellets, radial temperature gradients and their accompanying thermal stress fields are established in the pellets. In evaluating the acceptability of advanced designs utilizing boron carbide, thermal stress levels in pellets under irradiation have been predicted. Laboratory experimentation and fracture mechanics analysis have been used to characterize the type of fracture behavior to be anticipated.

- 1500 (HEDL-SA--1387) CLEANING, DISASSEMBLY, AND REQUALIFICATION OF THE FFTF IN VESSEL HANDLING MACHINE. Coops, W.J. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1977. Contract EY-76-C-14-2170. 26p. (CONF-7710126--1). Dep. NTIS, PC A03/MF A01.

From American power conference; Chicago, IL, USA (Oct 1977).

The Engineering Model In Vessel Handling Machine (IVHM) was successfully removed, cleaned, disassembled, inspected, reassembled and reinstalled into the sodium test vessel at Richland, Washington. This was the first time in the United States a full size operational sodium wetted machine has been cleaned by the water vapor nitrogen process and requalified for operation. The work utilized an atmospheric control system during removal, a tank type water vapor nitrogen cleaning system and an open "hands on" disassembly and assembly stand. Results of the work indicate the tools, process and equipment are adequate for the non-radioactive maintenance sequence. Additionally, the work proves that a machine of this complexity can be successfully cleaned, maintained and re-used without the need to replace a large percentage of the sodium wetted parts.

- 1501 (HEDL-SA--1388) FAST FLUX TEST FACILITY PRIMARY SODIUM VALVES. Rabe, G.B.; Ezra, B.C. (Foster Wheeler Energy Corp., Livingston, NJ (USA)). 1977. Contract EY-76-C-14-2170. 42p. (CONF-771237--1). Dep. NTIS, PC A03/MF A01.

From 1. joint US/Japan seminar on breeder reactor plant components; Los Angeles, CA, USA (5 Dec 1977).

The design and development of the valves used in the primary sodium coolant loop of the Fast Flux Test Facility is described. One tilting-disk check valve is used in the cold leg of the coolant loop. It is designed to limit flow reversal in the loop while maintaining a low pressure drop during forward flow. Two isolation valves are used in each

coolant loop--one in the cold leg and one in the hot leg. They are of the motor-operated swinging-gate type. The design, analysis, and testing programs undertaken to develop and qualify these valves are described.

- 1502 (HEDL-SA--1391) FRICTION, WEAR, AND CORROSION OF LAVES-HARDENED NICKEL ALLOY HARDSURFACING IN SODIUM. Johnson, R.N.; Farwick, D.G. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 23p. (CONF-780430--6). Dep. NTIS, PC A02/MF A01.

From Conference on metallurgical coatings; San Francisco, CA, USA (3 Apr 1978).

This paper represents the results of friction, wear, and corrosion tests on Tribaloy 700 specimens. The Tribaloy alloys are a family of materials typically consisting of a hard, intermetallic Laves phase dispersed in a cobalt or nickel eutectic or solid solution matrix. Only one alloy of the Tribaloy series, Tribaloy 700, is a cobalt-free nickel-base alloy. Tribaloy 700 is therefore of interest for nuclear reactor applications, where cobalt must be restricted. Tribaloy 700 rubbing against itself in sodium exhibited among the lowest friction coefficients measured for metallic materials at high temperature. Detonation-gun coatings had lower friction and lower corrosion rates than plasma coatings, in general. The wear rate of Tribaloy 700 was negligible. Surface damage resistance was good, especially when in contact with itself or other hard materials. The material also exhibited low corrosion rates in sodium.

- 1503 (HEDL-SA--1393) BIASES FOR CURRENT FFTF CALCULATIONAL METHODS. Ombrellaro, P.A.; Bennett, R.A.; Daughtry, J.W.; Dobbin, K.D.; Harris, R.A.; Nelson, J.V.; Peterson, R.E.; Rothrock, R.B. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 9p. (CONF-780401--9). Dep. NTIS, PC A02/MF A01.

From Topical meeting on advances in reactor physics; Gatlinburg, TN, USA (9 Apr 1978).

Uncertainties in nuclear data and approximate calculational methods used in safety design, and operational support of a reactor yield biased as well as uncertain results. Experimentally based biases for use in Fast Flux Test Facility (FFTF) core calculations have been evaluated and are presented together with a description of calculational methods. Experimental data for these evaluations were obtained from an Engineering Mockup Critical (EMC) of the FFTF core built at the Argonne National Laboratory (ANL). The experiments were conceived and planned by the Hanford Engineering Development Laboratory (HEDL) in cooperation with the Westinghouse Advanced Reactors Division (WARD) and ANL personnel, and carried out by the ANL staff. All experiments were designed specifically to provide data for evaluation of current FFTF core calculational methods. These comprehensive experiments were designed to allow simultaneous evaluations of biases and uncertainties in calculated reactivities, fuel sub-assembly and material reactivity worths, small sample worths, absorber rod worths, spatial fission rate distributions, power tilting effects and spatial neutron spectra. Modified source multiplication and reactivity anomaly methods have also been evaluated. Uncertainties in the biases have been established and are sufficiently small to attain a high degree of confidence in the design, safety and operational aspects of the FFTF core.

- 1504 (HEDL-SA--1399-FP) FABRICATION OF ThO_2 , UO_2 , AND PuO_2 - UO_2 PELLETS. Rasmussen, D.E.; Jentzen, W.R.; McCord, R.B. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 22p. (CONF-780545--10). Dep. NTIS, PC A02/MF A01.

From Meeting of the American Ceramic Society; Detroit, MI, USA (6 May 1978).

Portions of document are illegible.

Fabrication of ThO pellets for EBR-II irradiation testing and fabrication of UO_2 and PuO_2 - UO_2 pellets for United Kingdom Prototype Fast Reactor (PFR) irradiation testing is discussed. Effect of process parameters on density and microstructure of pellets fabricated by the cold press and sinter technique is reviewed.

- 1505 (HEDL-SA--1402) HIGH TEMPERATURE IN-REACTOR CREEP OF 20% COLD WORKED FTR CLADDING. Gilbert, E.R.; Lovell, A.J.; Chin, B.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 15p. (CONF-780722--24). Dep. NTIS, PC A02/MF A01.

From 9. symposium on effects of radiation in structural materials; Richland, WA, USA (10 Jul 1978).

Portions of document are illegible.

The in-reactor creep behavior of 20% cold worked Type 316 stainless steel tubing was investigated at temperatures ranging from 410 to 720°C for neutron fluences up to 5×10^{22} n/cm² ($E > 0.1$ MeV) in the experimental breeder reactor (EBR-II). The irradiation environment was found to either retard or enhance creep rates depending upon the irradiation conditions. When compared to thermal control tests on unirradiated specimens, irradiation enhanced creep rates were observed for temperatures below 620°C. Above 620°C, however, some specimens exhibited less in-reactor creep than thermal creep of unirradiated specimens. This irradiation hardening was observed for strain rates exceeding $5 \times 10^{-4}\%$ per hour. In-reactor and thermal creep rates were found to be similar at lower strain rates. To fully describe the experimentally observed in-reactor creep behavior with an empirical expression, a commutative equation was developed. The development was based as much as possible on existing equations for thermal creep and irradiation creep. The thermal creep description is comprised of a slow transient, rapid transient, and a tertiary creep term. The tertiary creep term which becomes important for strain rates greater than $5 \times 10^{-4}\%$ per hour has been modified to predict the irradiation hardening phenomena. Similarly, the irradiation-enhanced contributions to in-reactor creep are comprised of a primary irradiation-enhanced creep term, steady-state irradiation enhanced creep term, and a tertiary irradiation enhanced creep term.

- 1506 (HEDL-SA--1405) COMPARISON OF FUEL PIN DEFORMATIONS WITH PRESSURIZED TUBE CREEP TESTS. Taylor, J.P.; Gilbert, E.R.; Lovell, A.J. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1978. Contract EY-76-C-14-2170. 27p. (CONF-780722--7). Dep. NTIS (US Sales Only), PC A03/MF A01.

From 9. symposium on effects of radiation in structural materials; Richland, WA, USA (10 Jul 1978).

Fuel pins are a key component in the Fast Breeder Reactor; and the economical performance of the reactor depends upon the behavior and life of fuel pin assemblies. One of the limiting phenomena in fuel pin and assembly

performance is deformation. A thermo-mechanical performance code 'SIFAIL' is being used at HEDL to predict fuel cladding deformation. SIFAIL utilizes empirical descriptions of cladding strain as a function of temperature, stress and neutron fluence to predict fuel pin profiles. The equations are derived primarily from in-reactor tests on tubes pressurized with an inert gas and irradiated under conditions of constant temperature and stress. These equations were modified in SIFAIL to account for the variations in temperature and stress with time which occur in fuel pins. It is the purpose of these investigations to develop analytical methods which can be used to design fuel pins with optimum long life performance. Equations were developed to describe the in-reactor creep behavior for two different classes of AISI 316 stainless steel tubing; a developmental class for the FTR and the FTR first code tubing. The behavior of these two tubing classes were sufficiently different that two distinctly separate equations were required to describe their in-reactor creep behaviors.

- 1507 (HEDL-SA--1407) OUT-OF-REACTOR TRANSIENT FISSION GAS RELEASE STUDIES. Hinman, C.A.; Slagle, O.D. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 19p. (CONF-780545--7). Dep. NTIS, PC A02/MF A01.

From Meeting of the American Ceramic Society; Detroit, MI, USA (6 May 1978).

The work described is a series of six laboratory thermal transient tests using fast neutron irradiated mixed oxide fuel material. These tests, called FGR tests for Fission Gas Release, were all performed on fuel sections cut from the fuel pin PNL-2-4, which had been irradiated in EBR-II. This pin was irradiated to an average burnup of 3.6 atom percent at 250 watts/cm. The objective of the tests was to determine the transient interactions of the fuel and retained fission gas under thermal conditions resembling a loss of sodium coolant flow type of hypothetical accident in a fast reactor. These data help provide the experimental base for the development of computer codes for modeling hypothetical reactor accidents.

- 1508 (HEDL-SA--1420) EFFECT OF SODIUM ENVIRONMENT ON BREEDER REACTOR COMPONENTS. Eichelberger, R.L.; Brehm, W.F. (Liquid Metal Engineering Center, Canoga Park, CA (USA); Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 39p. (CONF-780308--8). Dep. NTIS, PC A03/MF A01.

From NACE meeting on corrosion; Houston, TX, USA (6 Mar 1978).

This paper reviews the interaction phenomena between sodium and reactor materials, and the application to reactor design and operation. Both corrosion and deposition are considered. Sodium removal and decontamination is briefly mentioned, together with some recent developments that are still in the experimental stage. No new theories of sodium-material interaction behavior are offered. 56 references.

- 1509 (HEDL-SA--1423) FUEL PIN BUNDLE AND HEXAGONAL DUCT MECHANICAL INTERACTION. Chan, D.P.; Jackson, R.J. (Westinghouse Hanford Co., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 24p. (CONF-780609--13). Dep. NTIS, PC A02/MF A01.

From ASME/CSME pressure vessels and piping conference; Montreal, Canada (25 Jun 1978).

Cause and effect of the fuel pin bundle and the hexagonal duct mechanical interaction are discussed. An analytical method for calculating the bundle compressive stiffness and the pin-to-pin clearance is introduced. Analytical results for bundles of different sizes are compared to the available experimental values. It is concluded that (1) the bundle/duct interactive force and reduction in pin-to-pin clearance can vary over wide ranges due to tolerances on wire tension and wire pitch, (2) the maximum contact force and the maximum reduction in pin-to-pin clearance are more likely to occur near the edge pin than in the interior of a bundle, and (3) FFTF pin bundle will not experience adverse effects of bundle duct mechanical interaction.

- 1510 (HEDL-SA--1425) FLOW IMPEDANCE TEST RESULTS AT HEDL: CHEMICAL BASIS. Yunker, W. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1978. Contract EY-76-C-14-2170. 23p. (CONF-780677--6). Dep. NTIS, PC A02/MF A01.

From U.S. Japanese information exchange; Richland, WA, USA (Jun 1978).

In stainless steel liquid sodium heat transfer systems, the steel constituents undergo mass transfer. Transported material collects on the heat transfer surfaces in the form of films or crystals. The deposits usually are found in the cooler regions of the system where the temperature may be too low for significant diffusion of these elements into the base steel. Deposits which break free can add to the particulate inventory of the system. Materials reported included α -Fe, NaCrO_2 , M_2C_3 , Cr_2C_6 , Mn and Cr_2C . The loss of material from the high temperature regions can result in a permanently roughened surface after extended periods of time, especially if the oxygen activity in the sodium has been high. The element silicon (1 w/o in stainless steel) has been ignored generally as being significant in mass transfer and sodium impurity control. The tests described show that, quite to the contrary, silicon is the cause of a number of flow, heat transfer and operational problems in such systems. Sodium tests have been run at HEDL to determine the cause of an apparent increase in friction factor which was observed during the testing of a 217-pin bundle in a 590°C isothermal sodium system.

- 1511 (HEDL-SA--1429FP) SUMMARY OF SODIUM REMOVAL AND DECONTAMINATION PROGRAMS IN THE U.S.A. Steele, D.P. III; Brehm, W.F. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1978. Contract EY-76-C-14-2170. 8p. (CONF-780232--3). Dep. NTIS, PC A02/MF A01.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

The goals of the United States Department of Energy sodium removal and decontamination programs are (1) to identify plant requirements and develop safe effective processes for sodium removal, both for experimental facilities and for reactor components, (2) to develop effective decontamination processes for removing deposited radioactivity, (3) to establish and put into use a set of workable criteria for requalification and return to service of components after sodium removal and decontamination, (4) to design, build and operate facilities in which to perform the sodium removal and decontamination operations. The paper gives a summary of progress toward each of these goals. Three basic processes for sodium removal have been investigated extensively: the use of water vapor in an inert carrier gas, the use of alcohol-type reagent,

and evaporation. The process development work on the first two processes has been essentially completed. The evaporative process is still under development, but preliminary parameters have been established.

- 1512 (HEDL-SA--1432) SODIUM SAMPLING AND IMPURITY MONITORING SYSTEMS IN U.S. BREEDER REACTORS. McCown, J.J.; Olson, W.H. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 30p. (CONF-780677--4). Dep. NTIS, PC A03/MF A01.

From U.S. Japanese information exchange; Richland, WA, USA (Jun 1978).

In the U.S. Breeder Reactor Program, sodium samplers and on-line impurity monitoring devices are used to provide complete sodium characterization data. The EBR-II has had such a program in place for over 10 years following the behavior of all detectable metals, non-metals and radionuclides. FFTF will be equipped with samplers and monitors which will be used to characterize all its sodium systems. This program will start in the summer of 1978 during sodium fill operations and continue for the life of the plant.

- 1513 (HEDL-SA--1433) FUEL FAILURE MONITORING SYSTEMS IN U.S. BREEDER REACTORS. McCown, J.J.; Lambert, J.D.B. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 30p. (CONF-780677--3). Dep. NTIS, PC A03/MF A01.

From U.S. Japanese information exchange; Richland, WA, USA (Jun 1978).

In the U.S. Breeder Reactor Program, sodium cooled test reactors are equipped with systems for detecting, characterizing and locating fuel cladding breaches. Gas releases and sodium-fuel contact events are analyzed on EBR-II by continuous on-line cover gas and sodium monitoring techniques and by the laboratory analysis of periodic samples. On FFTF hardware, electronics and computer software required for FFM have been developed, designed and tested and are in place awaiting plant startup.

- 1514 (HEDL-SA--1439) HELIUM PRODUCTION IN FBR OUT-OF-CORE STRUCTURAL COMPONENTS. Simons, R.L. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 22p. (CONF-780722--14). Dep. NTIS, PC A02/MF A01.

From 9. symposium on effects of radiation in structural materials; Richland, WA, USA (10 Jul 1978).

The major sources of helium production in stainless steel irradiated in a fast reactor are (n, α) reactions in nickel, iron, chromium, and impurities such as nitrogen and boron. The corresponding reaction cross sections are generally of the threshold type with threshold energies above one MeV. Two important exceptions are the cross sections for the two-stage nickel reaction and the boron 10 reaction. Although lifetime helium concentrations are low, these concentrations may be an important consideration when applying material property change data obtained from the EBR-II core to high temperature out-of-core EBR structural components. The ratio of appm helium to displacements per atom is larger for the soft out-of-core spectra than for the harder in-core spectra in which most irradiation data are obtained. These differences should be kept in mind when applying data obtained in EBR-II to FBR out-of-core components.

1515 (HEDL-SA--1440) DELAYED NEUTRON PARAMETERS FOR THE FFTF. Daughtry, J.W.; Dobbin, K.D. (Westinghouse Hanford Co., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 13p. (CONF-780622--48). Dep. NTIS (US Sales Only), PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

Delayed neutron parameters have been calculated for the Fast Test Reactor (FTR) and for the FTR Engineering Mockup Critical (EMC) assembly, using the most recent available evaluations of delayed neutron data. Using ENDF/B Version 4 delayed neutron parameters, β /sub eff/ is about 4% larger than the previously accepted value and the inhour-to-percent $\Delta k/k$ conversion factor is about 4% smaller. For beginning-of-life conditions, ^{235}U , ^{240}Pu , and ^{241}Pu contribute less than 10% to the total β /sub eff/ for the FTR and EMC, and, therefore, uncertainties in their delayed neutron data are relatively unimportant. Kinetics parameters calculated with a two-dimensional model of the FTR are less than a percent different from those obtained with a three-dimensional calculation. Thus the less expensive two-dimensional calculations are adequate.

1516 (HEDL-SA--1441) HANDLING OF SODIUM FOR THE FFTF. Ballif, J.L.; Meadows, G.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1978. Contract EY-76-C-06-1830. 15p. Dep. NTIS, MF A01.

Portions of document are illegible.

Based on the High Temperature Sodium Facility (HTSF) experience and the extensive design efforts for FFTF, procedures are in place for the unloading of the tank cars and for the fill of the FFTF reactor. Special precautions have been taken to provide safe handling and to accommodate contingencies in operation. These contingencies include special protective suits allowing personnel to enter and correct conditions arising from fill operations in the course of moving 7.71×10^5 kg (1.7×10^6 lbs) of sodium from the tank cars into the reactor vessel and its loop system.

1517 (HEDL-SA--1442) LIQUID METAL LEVEL MEASUREMENT SYSTEMS FOR REACTOR SERVICE. Cota, L.; Connell, J.W. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 56p. (CONF-780677--1). Dep. NTIS, PC A04/MF A01.

From U.S. Japanese information exchange; Richland, WA, USA (Jun 1978).

A program was initiated by the U.S. Government in 1971 to test the available liquid metal level instruments on the market to determine which ones could be used in the Liquid Metal Breeder Reactor Program. The testing agency purchased the available level instruments on the market and started a testing program. The test results from that testing program are included in the report. The testing program is continuing.

1518 (HEDL-SA--1444) WATER VAPOR NITROGEN PROCESS FOR REMOVING SODIUM FROM LMFBR COMPONENTS. Crippen, M.D.; Funk, C.W.; Lutton, J.M. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1978. Contract EY-76-C-14-2170. 8p. (CONF-780232--2). Dep. NTIS (US Sales Only), PC A02/MF A01.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

Application and operation of the Water Vapor-Nitrogen Process for removing sodium from LMFBR components is reviewed. Emphasis is placed on recent efforts to verify the technological

bases of the process, to refine the values of process parameters, and to ensure the utility of the process for cleaning and requalifying components.

1519 (HEDL-SA--1446) RECENT WORK ON SODIUM-COOLED REACTOR PURIFICATION SYSTEMS. Bloom, G.R.; McPheeters, C.C. (Hanford Engineering Development Lab., Richland, WA (USA); Argonne National Lab., IL (USA)). Feb 1978. Contract EY-76-C-14-2170. 46p. (CONF-780677--2). Dep. NTIS, PC A03/MF A01.

From U.S. Japanese information exchange; Richland, WA, USA (Jun 1978).

Fast Flux Test Facility (FFTF) cold traps, including the Reactor Primary Cold Trap, the Reactor Secondary Cold Trap and the Closed Loop Cold Trap were fabricated and their thermal performances measured on sodium test systems. The cold traps all met reactor requirements after necessary economizer modifications. Mercury tests were performed to study the effect of woven wire packing on the flow of a liquid metal adjacent to a heated vertical wall as occurs in the cold trap crystallizer. The mesh was observed to alter the velocity distribution in the natural convection flow region. Cold trapping experiments have been done to determine the interaction of hydrogen and tritium isotopes during precipitation. It was determined that the hydrogen removal rate is of primary importance in tritium removal from sodium. Isotopic exchange plays a negligible role in cold trapping tritium. A one-dimensional computer model has been developed to predict hydride and oxide precipitation patterns. Experiments have been done using an essentially one-dimensional (long, thin) tube to compare results with the model.

1520 (HEDL-SA--1447-8) FUEL COLUMN STABILITY IN IRRADIATED MIXED-OXIDE FUEL. Melde, G.F.; Weber, J.W.; Dutt, D.S. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1978. Contract EY-76-C-14-2170. 7p. (CONF-780622--58). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

The dimensional stability of the fuel columns of 250 mixed uranium-plutonium-oxide fuel pins irradiated in EBR-II was evaluated including length and fuel-to-cladding gap size changes. Fuels with spherical porosity and fuel with nonspherical porosity were included in the study. Sufficient insight into the basic processes responsible for fuel behavior has been gained from this study to allow evaluation of expected behavior of fuel in advance of irradiation. All evidence indicates that mixed-oxide fuel columns for fast reactor applications are dimensionally stable and should present no problem to reactor operations.

1521 (HEDL-SA--1448) MASS AND ACTIVITY TRANSPORT IN FLOWING SODIUM SYSTEMS. Brehm, W.F.; Colburn, R.P.; McGuire, J.C.; Anantamula, R.P.; Chulos, L.E.; Maffei, H.P.; White, P.A. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 14p. (CONF-780493--1). Dep. NTIS, PC A02/MF A01.

From U.S. Japanese specialists meeting; Argonne, IL, USA (17 Apr 1978).

The objective of the studies described is to develop and demonstrate methods for reducing the transport of radioactive species from breeder reactor core regions to maintenance zones adjacent to primary heat transport system (PHTS) piping and equipment. Both corrosion

products (^{54}Mn , ^{60}Co) and fission products (^{137}Cs , $^{140}\text{Ba-La}$, tritium) are considered.

- 1522 (HEDL-SA--1449) CALCULATION OF RESONANT REACTION RATES NEAR REGION INTERFACES IN THE FFTF. Rothrock, R.B. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 13p. (CONF-780622--57). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

Proper prediction of the local Pu-239 production, power generation, and temperatures reached in experimental U-238 blanket assemblies represents a problem in the calculation of spatial resonant reaction rate distributions not addressed in the course of FFTF nuclear design. These experimental assemblies generally have a common boundary with the FTR axial reflector region and may also, depending on their desired irradiation position in the core, be located adjacent to the FTR radial reflector. At these interfaces, the reflected neutron current from the axial or radial reflectors can cause a significant local increase in the capture rate. This is an effect not accounted for in the standard FTR multigroup diffusion theory calculations using the shielding factor method for preparation of broad group cross sections for resonant isotopes. Consequently, a calculational procedure was developed to treat explicitly the local variations in U-238 capture rate in these situations.

- 1523 (HEDL-SA--1452) CHARACTERIZATION OF CORROSION PRODUCT DEPOSITS IN SODIUM SYSTEMS. Colburn, R.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1978. Contract EY-76-C-14-2170. 17p. (CONF-780232--4). Dep. NTIS, PC A02/MF A01.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

Portions of document are illegible.

The characterization of mass transfer deposits on the containment surface of sodium systems is an essential step in the development of LMFBR decontamination processes. The surfaces of a sodium test facility used to study radioactive mass transport were examined to determine the chemical and physical nature of the deposits. The examination showed two fundamentally different problems will occur in the hot and cold leg regions of an LMFBR heat transport system. In the hot leg ^{60}Co , ^{54}Mn and ^{59}Fe will diffuse into the base metal and decontamination may require the removal of several microns of the base metal. In the cold leg ^{54}Mn will be the dominant radionuclide. It will be largely incorporated in nickel-manganese rich particles, a few microns in diameter, scattered on the surface as a semiaherent deposit with very shallow penetration into the base metal. For this region a process should be developed which selectively attacks the deposit phase with minimal attack on the base metal.

- 1524 (HEDL-SA--1453) THREE-DIMENSIONAL INELASTIC ANALYSIS OF A SMALL LIQUID SODIUM VALVE. Weiner, E.O. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 29p. (CONF-780609--10). Dep. NTIS, PC A03/MF A01.

From ASME/CSME pressure vessels and piping conference; Montreal, Canada (25 Jun 1978).

A three-dimensional inelastic analysis of a small liquid sodium valve for the Fast Flux Test Facility was conducted to assess

ratcheting and creep damage at temperatures to 1200°F using the MARC-CDC finite element program. A single mesh of twenty-node solid elements, representing the Y-type cylinder intersection, is used for both the heat transfer and mechanical computer runs. Material behavior is governed by kinematic and creep strain hardening, and the loading consists of steady mechanical loads with cyclic fluid thermal transients. The resulting creep damage is significantly smaller than that of a corresponding axisymmetric analysis, and plastic strains are confined to a single region.

- 1525 (HEDL-SA--1455) EFFECTS OF IRRADIATION CREEP ON EX-REACTOR MECHANICAL PROPERTIES. Duncan, D.R. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1978. Contract EY-76-C-14-2170. 23p. (CONF-780722--6). Dep. NTIS (US Sales Only), PC A02/MF A01.

From 9. symposium on effects of radiation in structural materials; Richland, WA, USA (10 Jul 1978).

Irradiation creep of reactor components is usually considered nondamaging. This nondamaging concept has had no experimental verification; therefore, confidence in its application to reactor design is reduced. The damaging nature of irradiation creep was tested in the work described. Material used was SS-316 cladding of FFTF geometry, 5.84 mm x 5.08 mm, which had been irradiated under stress to provide significant amounts of irradiation creep. Specimens with no irradiation or stress exposure and specimens with zero stress irradiation exposure were used as controls. The actual test mode was a transient test, which imposes a constant internal stress on a closed piece of tubing and superimposes an increasing constant-rate temperature ramp on the tubing. The tests were conducted until failure occurred by tubing breach. Measured parameters were failure temperature and strain at failure. The effect of irradiation creep was monitored by comparing failure temperature and strain of specimens that had irradiation creep with failure parameters of specimens that had no irradiation creep. Results showed that specimens with irradiation creep failed at temperatures similar to those without irradiation creep. Strain results were less clear, but no major strain reductions were seen; thus, irradiation creep to the magnitudes investigated is nondamaging.

- 1526 (HEDL-SA--1458) PLANNED NATURAL CIRCULATION TESTING IN THE FAST FLUX TEST FACILITY. Bouchey, G.D.; Additon, S.L.; Parziale, E.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1978. Contract EY-76-C-14-2170. 31p. (CONF-780622--51). Dep. NTIS, PC A03/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

The natural circulation testing of FFTF is planned during three distinct periods in the Plant Acceptance Testing Program: secondary loop testing; low power (<5%) testing in the primary and secondary loops; and, high power (greater than or equal to 5%) transient testing. Planned testing during each of these periods and some typical test predictions that have been completed to-date will be discussed in more detail below.

- 1527 (HEDL-SA--1459) FFTF HOT LEG RUPTURES AND LOSS OF FLOW EVENTS. Additon, S.L.; Chien, Y.M. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1978.

Contract EY-76-C-14-2170. 52p. (CONF-780622-46). Dep. NTIS, PC A04/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

Two examples of recent FFTF design studies are discussed to illustrate the level of analytic complexity required to resolve design questions and the importance of system simulation. The first example is a study of postulated breaks in the primary hot leg piping. The second example is a study of the performance of the Plant Protection Systems during loss-of-flow events. Adequate safety margins were determined in both studies.

1528 (HEDL-SA--1462) TENSILE PROPERTY CORRELATIONS FOR HIGHLY IRRADIATED 20% C.W. TYPE 316 STAINLESS STEEL. Fish, R.L.; Cannon, N.S.; Wire, G.L. (Hanford Engineering Development Lab., Richland, WA (USA)). 11 May 1978. Contract EY-76-C-14-2170. 26p. (CONF-780722--3). Dep. NTIS, PC A03/MF A01.

From 9. symposium on effects of radiation in structural materials; Richland, WA, USA (10 Jul 1978).

Experiments on developmental FFTF cladding (20% C.W. Type 316 stainless steel) extended the data base to a fast neutron fluence of 8.4×10^{22} n/cm² ($E > 0.1$ MeV). The specimens were irradiated at temperatures ranging from 371° to 816°C, although peak fluence levels were attained on specimens irradiated near 371°C and 649°C. Tension tests were performed at 232°C, near the irradiation temperature, and in some cases, above the irradiation temperature. Test specimen strain rates ranged from 4×10^{-5} /s to 4×10^{-2} /s. Data generated on cladding irradiated near 371°C established that the low temperature strength and ductility are fluence independent beyond about 5×10^{22} n/cm² ($E > 0.1$ MeV). Strength behavior of the irradiated cladding at 538°, 593°, and 649°C is essentially the same as exhibited by thermally aged developmental cladding at the same temperatures and times. Up to a fluence of approximately 5×10^{22} n/cm² ($E > 0.1$ MeV), the 538°C ductility values remain relatively fluence independent after an initial decrease. Higher temperature (593°C and 649°C) ductilities decrease continually with increasing fluence. Tensile parameter correlations were developed for the prediction of irradiation effects on the tensile properties of 20% C.W. Type 316 stainless steel. These correlations are based on unirradiated tensile property correlations developed using Hart's equation of state. The condition of plastic deformation of materials such as 316 stainless steel, can be characterized by a structure parameter (σ^*) which describes the material's "hardness." Irradiation effects can be incorporated into this formulation by parameterizing the changes in σ^* with irradiation temperature and fluence. Correlations provide a description of strength and ductility over the temperature range of 371°C to 871°C and strain rates of 10^{-5} to 10^1 /s.

1529 (HEDL-SA--1463FP) EFFECT OF POWER SKEW ON THE FTR TOP. Yung, S.C.; Wilburn, N.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1978. Contract EY-76-C-14-2170. 36p. (CONF-780622--55). Dep. NTIS, PC A03/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

An enduring program of studies, both experimentally and analytically, on the unprotected Transient Overpower Hypothetical Core Disruptive Accident (TOP-HCDA) of the Liquid Metal Fast Breeder Reactor (LMFBR) has

been in progress for a decade or so. The data available from the experiments and the computational methodology improvements have made the recent TOP safety assessments more realistic. The evidences from the in-pile tests in the TREAT facility, such as the axial location of fuel pins' failure towards the upper portion of the pin, and the rather benign nature of Molten Fuel and Coolant Interactions (MFCI) have unambiguously clarified some key issues of earlier TOP accident assessments. However, on the other hand, there is no conclusive assertion that can be made on the post-failure scenarios. No complete sweepout of molten fuel has ever been observed in all TREAT TOP tests performed to date to warrant the complete sweepout assumption in the TOP analysis. As a matter of fact, the fuel blockage has occurred in almost all TREAT TOP tests, at least partial blockage, in the coolant flow paths in the testing apparatus. Arguments on the validity of these TREAT TOP experiments to Fast Test Reactors (FTR) have been raised because of the low coolant flow velocity and low coolant pressure drops in the testing section in their hydraulic simulations and gross power overdriving after fuel pin failure. For the non-prototypicality of hydraulic simulation, it argues that the higher coolant velocity will increase the likelihood of hydraulic fuel sweepout and the higher the system hydraulic pressure will reduce the likelihood of coolant flow reverse. A scrutiny of the hydraulic consequences may reveal that this hydraulic non-prototypicality does not necessarily yield a favorable conclusion to the formation of blockage in the FTR.

1530 (HEDL-SA--1469) FATIGUE-CRACK PROPAGATION ANALYSIS OF LMFBR PIPING. James, L.A. (Westinghouse Hanford Co., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 15p. (CONF-780609--11). Dep. NTIS, PC A02/MF A01.

From ASME/CSME pressure vessels and piping conference; Montreal, Canada (25 Jun 1978).

The techniques of linear-elastic fracture mechanics were employed to estimate the in-service extension of hypothetical flaws in two LMFBR primary piping elbows. The paper describes the technical basis for the conclusion that very little crack growth will occur in these components due to either the anticipated normal, upset, and emergency events, as well as some unlikely "abnormal" events. The two examples treated include a hot leg elbow and a cold leg elbow. The assumptions and approximations concerning stress and loading conditions, the expected in-service operational environment, and material property behavior are discussed in detail. Based upon the analyses, it is concluded that significant crack extension is very unlikely under these LMFBR primary piping conditions.

1531 (HEDL-SA--1470-S) EFFECTS OF STOICHIOMETRY ON CLADDING REACTION IN MIXED-OXIDE FUEL AT HIGH BURNUP. Lawrence, L.A.; Weber, J.W.; Devary, J.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1978. Contract EY-76-C-14-2170. 7p. (CONF-780622--76). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

The effects of initial stoichiometry on the character and extent of fuel-cladding chemical interaction (FCCI) for mixed-oxide fuels at O/M's of 1.984 and 1.966 irradiated in EBR-2 to peak burnup are described. The strong effect of initial O/M character and extent of FCCI persists to high burnup. Wastage correlations

for mixed-oxide fuel based on data from higher O/M fuel must be reduced accordingly to account for the lower O/M levels typical of current LMFBR fuel designs. (FS)

- 1532 (HEDL-SA--1477) FBR CORE DESIGN WITH THE COMPOSITE FUEL ASSEMBLY. Cappiello, M.W. (Hanford Engineering Development Lab., Richland, WA (USA)). [nd]. Contract EY-76-C-14-2170. 18p. (CONF-780622--62). Dep. NTIS, PC A02/MF A01.
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

Although calculations are preliminary, overall feasibility of an FBR core design with the composite fuel assembly has been demonstrated. The advantaged over the heterogeneous design is that large variances in assembly mixed mean outlet temperatures are eliminated. Also, the effective enrichment of an assembly may easily be adjusted by varying the number of fertile pins per assembly, thus making it possible to flatten the core radial power profile. The use of the composite fuel assembly may in the future offer a significant alternative to heterogeneous FBR core design.

- 1533 (HEDL-SA--1485) IRRADIATION TEST PROGRAM FOR FFTF. Corrigan, D.C.; Last, G.A. (Hanford Engineering Development Lab., Richland, WA (USA)). [nd]. Contract EY-76-C-14-2170. 10p. (CONF-780622--63). Dep. NTIS, PC A02/MF A01.
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

Four unique design features are described which make the Fast Flux Test Facility eminently suitable for irradiation test programs. These features are a fast flux level of 7×10^{15} neutrons/cm²/sec, a 36-inch reference (breeder reactor) core height, test volumes suitable for testing of statistical quantities of materials, and the capability for direct (contact) or indirect (proximity) instrumentation of active core experiments.

- 1534 (HEDL-SA--1489-FP) APPLICABILITY OF TREAT TOP EXPERIMENTS TO POST-FAILURE DYNAMICS. Padilla, A. Jr.; Smith, D.E. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1978. Contract EY-76-C-14-2170. 23p. (CONF-780622--69). Dep. NTIS, PC A02/MF A01.
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

Recent TREAT transient overpower experiments have provided valid data to evaluate the post-failure phenomenology associated with LMFBR transient overpower accidents. Some comments are made on the role of the TREAT experiments and the experiment analysis.

- 1535 (HEDL-SA--1490) FFTF AND THE ASME CODE. Tobin, J.C. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract WY-76-C-14-2170. 37p. (CONF-780240--1). Dep. NTIS, PC A03/MF A01.
From Meeting of Washington state boiler inspectors; Seattle, WA, USA (27 Feb 1978).
Photographs are presented of the FFTF reactor facility, components, and some materials.

- 1536 (HEDL-SA--1494) EXPERIMENTAL EVALUATION OF CELL LINERS. Wierman, R.W.; Simmons, L.D.; Muhlestein, L.D. (Hanford Engineering Development Lab., Richland, WA (USA)). [nd]. Contract EY-76-C-14-2170. 28p. Dep. NTIS, PC A03/MF A01.

Cell liners may be used in breeder reactor sodium pipe ways, sodium cells, and lower cavity region to provide a leak-tight cell and to protect the concrete from sodium in the unlikely event of a sodium spill. The objectives of the HEDL liner verification test program are to evaluate the integrity of liner concepts under postulated accident conditions and to develop the experimental data base which will demonstrate that liners will not fail. Two specific tests are reported; a high temperature liner feature test, and a large-scale liner sodium spill test. In both tests no failures of the liners or tendencies toward failure were detected. The discussed liner designs appeared to be conservative, and the liner strength appeared to be more than adequate.

- 1537 (HEDL-SA--1496) EXPERIENCE IN THE DESIGN, CONSTRUCTION, AND TESTING OF THE FFTF HEAT TRANSPORT SYSTEM. Dowling, R.J.; Ferguson, R.L.; Weber, S.A. (Department of Energy, Washington, DC (USA); Department of Energy, Richland, WA (USA). Richland Operations Office; Westinghouse Hanford Co., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 9p. (CONF-780407--3; IAEA-SM--225/22). Dep. NTIS, PC A02/MF A01.

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The Fast Flux Test Facility (FFTF) at Richland Washington, U.S.A., is an integrated plant complex that will provide the testing capability essential to the diverse needs of the breeder reactor program. The design, development and testing of the heat transport system components are described. This includes the sodium pumps, intermediate heat exchangers and dump heat exchangers for both the main plant and also for the closed loops. Using models and prototypes, each component has been multi-phase tested, modified when necessary, and then retested before plant installation. Simplified stress analysis methods for piping operating over 800°F (425°C) are noted. Plant construction is essentially complete and the preoperational tests are well underway. A description of the irradiation services for the FFTF is included.

- 1538 (HEDL-SA--1499) APPLICATION OF QUALITY ASSURANCE TO THE DESIGN, PROCUREMENT, AND CONSTRUCTION OF THE FAST FLUX TEST FACILITY. Garland, D.L.; Glasscock, R.B.; Pressesky, A.J. (Westinghouse Hanford Co., Richland, WA (USA); Department of Energy, Richland, WA (USA). Richland Operations Office; Department of Energy, Washington, DC (USA)). 1978. Contract EY-76-C-14-2170. 8p. (CONF-780407--3). Dep. NTIS, PC A02/MF A01.

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

Experience has clearly demonstrated the need for effective quality assurance programs throughout the design, procurement and construction phases of FFTF. Formal quality assurance programs for FFTF have contributed significantly to the achievement of project objectives thus far in a safe, reliable and predictable manner. Quality assurance programs employed by major FFTF Project participants are based in the requirements of RDT Standard F 2-2, one of the first and most comprehensive standards of its kind. RDT F 2-2 is similar in its basic requirements to ANSI N45.2 and other quality assurance program standards but differs in its degree of specificity, particularly in its coverage of development activities. RDT F 2-

2 is widely applied to a variety of large and small nuclear energy development projects and technology programs. Full implementation of RDT F 2-2 across the FFTF Project has had its problems and rewards. Problems have been encountered involving misinterpretation or overapplication of requirements. In some instances the problems pointed to the need for more specific requirements, and so the standard was amended to rectify the shortcomings. One valuable lesson learned is that feedback of use experience is essential to the development of a living, viable standard which can be beneficially applied to other projects. Benefits and cost savings have accrued from the preventive aspects of RDT F 2-2. Some examples of experiences and resulting amendments to RDT F 2-2, and their favorable impact on facility costs and schedule, are discussed in this paper.

- 1539 (HEDL-SA--1500-FP) FUEL PINS AND CORE RESPONSE UNDER LMFBR TOP ACCIDENT CONDITIONS. Wilburn, N.P.; Smith, D.E.; Baars, R.E.; Atcheson, D.B.; Spencer, B.W. (Hanford Engineering Development Lab., Richland, WA (USA); General Electric Co., Sunnyvale, CA (USA); Argonne National Lab., IL (USA)). 1978. Contract EY-76-C-14-2170. 18p. (CONF-781022--38). Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

Out-of-reactor experiments are currently being performed at Argonne National Laboratory to examine fuel sweepout and related post-failure phenomena under hypothetical TOP accident conditions. These tests are supplementing the TREAT MARK-II loop data base by keying on effects of important parameter variations such as system hydraulics and intrabundle coherency. In these tests, molten UO_2 , generated by a thermite reaction at 3470°K, is injected over approximately 40 msec into flowing sodium in a bundle of simulated LMFBR-type fuel pins. Hydraulic conditions in the bundle are selected to match conditions in either the MARK-II loop (HUMP-series) or the current design LMFBR subassembly (CAMEL-series). To date, four tests have been performed in both single-pin and seven-pin configurations representing coherent and incoherent subassembly power-to-flow cases, respectively. Details of the fuel motion were observed using a flash x-ray cine system. A compilation of significant findings from the four sweepout tests is presented.

- 1540 (HEDL-SA--1501-FP) HIGH TEMPERATURE PENETRANT PROCEDURE FOR LIQUID METAL PIPE WELDS. Hubbard, S.K.; Proudfoot, E.A. (Hanford Engineering Development Lab., Richland, WA (USA)). 9 May 1978. Contract EY-76-C-14-2170. 24p. (CONF-781008--4). Dep. NTIS, PC A02/MF A01.

From ASNT fall conference; Denver, CO, USA (2 Oct 1978).

A high temperature penetrant test developed for in-service inspection at the FFTF reactor can be performed at 400°F (205°C) achieving equal sensitivity to a standard test at ambient room temperature. However, the procedural control such as dwell time, cleaning, and development must be tightly controlled to achieve this result. The personnel performing the test must be well trained in the penetrant process and will require further special training in the high temperature technique.

- 1541 (HEDL-SA--1504-FP) EXPERIMENTAL CONFIRMATION OF THE DESIGN TO MINIMIZE VIBRATION AND WEAR IN 61-PIN WIRE-SPACED EBR-II

SUBASSEMBLIES. Fukuda, S.K. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1978. Contract EY-76-C-14-2170. 5p. (CONF-780622--71). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

Examinations of HEDL 61-pin subassemblies comprised of 5.84 mm (0.230) inch diameter mixed-oxide fuel pins with 1.02 mm (0.040") diameter spacer wire (PNL-9, -10, -11, HEDL-N-E, -N-F), showed severe cladding and spacer wire wear after irradiation in EBR-II. A comparison of a large number of design, fabrication, and irradiation parameters for all of the HEDL subassemblies indicated that the porosity per ring of fuel pins correlated significantly with the occurrence of wear on the fuel pins. The porosity per ring is the clearance between the flat-to-flat pin bundle dimension and the inner hex can dimension divided by the number of hexagonal fuel pin rings in the subassembly. The porosity per ring for PNL-9, -10, -11 and HEDL-N-E was 0.15 mm/ring (6 mils/ring) and 0.18 mm/ring (7 mils/ring) for the HEDL-N-F subassembly. Since the original FTR subassembly design had a porosity/ring spread of 0.04 mm/ring to 0.16 mm/ring (1.67 to 6.11 mils/ring) an additional series of irradiation tests was conducted to confirm that a tighter fuel pin bundle would eliminate the wear.

- 1542 (HEDL-SA--1505) OPTIMIZED FFTF ACCEPTANCE TEST PROGRAM COVERING PHASES III, IV, AND V. Wykoff, W.R.; Jones, D.H. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1977. Contract EY-76-C-14-2170. 84p. Dep. NTIS, PC A05/MF A01.

A detailed review of Phases III, IV, and V of the FFTF Acceptance Test Program has been completed. The purpose of this review was to formulate that test sequence which not only meets requirements for safe, reliable and useful operation of the plant, but also results in the earliest prudent demonstration of full-power performance. A test sequence based on the underlying assumption that sodium flows into the secondary sodium storage tank (T-44) no later than August 31, 1978, is described in detail. A time-scale which allows extra time to put systems and equipment into operation the first time, debugging, and learning how to operate most effectively has been superimposed on the test sequence. Time is not included for major equipment malfunctions. This test plan provides the basis for coordinating the many and varied activities and interfaces necessary for successful and timely execution of the FFTF Acceptance Test Program. In this report, the need dates have been identified for presently scheduled test articles and standard core components.

- 1543 (HEDL-SA--1506) DESIGN OF SODIUM COOLED REACTOR SYSTEMS AND COMPONENTS FOR MAINTAINABILITY. Carr, R.W.; Charnock, H.O.; McBride, J.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1978. Contract EY-76-C-14-2170. 43p. (CONF-7809124--1). Dep. NTIS, PC A03/MF A01.

From 1978 joint power generation conference; Dallas, TX, USA (Sep 1978).

Special maintainability problems associated with the design and operation of sodium cooled reactor plants are discussed. Some examples of both good and bad design practice are introduced from the design of the FFTF plant and other plants. Subjects include design for drainage, cleaning, decontamination, access, component removal, component disassembly and

reassembly, remote tooling, jigs, fixtures, and design for minimizing radiation exposure of maintenance personnel. Check lists are included.

- 1544 (HEDL-SA--1514) NEUTRON RADIOGRAPHY FACILITY AT THE HANFORD ENGINEERING DEVELOPMENT LABORATORY. Jackson, C.N. Jr.; Barton, J.P.; Proudfoot, E.A. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 25p. (CONF-781008--2). Dep. NTIS, PC A02/MF A01.

From ASNT fall conference; Denver, CO, USA (2 Oct 1978).

A 250 kW TRIGA reactor has recently been installed at the Hanford Engineering Development Laboratory (HEDL) to provide a neutron radiography facility for examinations of LMFBR fuel and other core components. The facility includes a vertical neutron beam for unirradiated materials and provisions for a horizontal beam which will be used for irradiated materials. Design of the horizontal beam is still being developed, whereas the vertical collimator equipment has been installed and is being applied and evaluated. Before the HEDL neutron radiography facility became available, a total of (1700) unirradiated driver fuel pins had been sent to the General Electric-Vallecitos Center for neutron radiography. The ability to provide valuable information concerning central voids within the fuel, inclusions between fuel pellet and clad, and fissile density variations caused by chemical reactions during manufacture, was demonstrated. These examinations served as a control on quality and also provided baseline data for future evaluations after FFTF irradiation. The paper describes the TRIGA reactor and gives details of the collimator and ancillary components installed for inspection of unirradiated fuel. Results of a testing program to determine neutron beam characteristics and identify radiographic quality of the images are discussed. Examples of production radiography and possible future applications are presented.

- 1545 (HEDL-SA--1516) ELECTRO-OPTICAL FUEL PIN IDENTIFICATION SYSTEM. Kirchner, T.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1978. Contract EY-76-C-14-2170. 13p. (CONF-780958--3). Dep. NTIS, PC A02/MF A01.

From 10. electro optics laser conference; Boston, MA, USA (19 Sep 1978).

A prototype Electro-Optical Fuel Pin Identification System referred to as the Fuel Pin Identification System (FPIS) has been developed by the Hanford Engineering Development Laboratory (HEDL) in support of the Fast Flux Test Facility (FFTF) presently under construction at HEDL. The system is designed to remotely read an alpha-numeric identification number that is roll stamped on the top of the fuel pin end cap. The prototype FPIS consists of four major subassemblies: optical read head, digital compression electronics, video display, and line printer.

- 1546 (HEDL-SA--1517) SODIUM VAPOR DEPOSITS OBSERVED AT HEDL ON LARGE COMPONENTS. Funk, C.W.; Eichelberger, R.L. (Hanford Engineering Development Lab., Richland, WA (USA); Atomic International Div., Canoga Park, CA (USA)). [nd]. Contract EY-76-C-14-2170. 18p. (CONF-780677--5). Dep. NTIS, PC A02/MF A01.

From U.S. Japanese information exchange; Richland, WA, USA (Jun 1978).

Testing of the Fast Flux Test Facility (FFTF) components in the High Temperature

Sodium Facility (HTSF) has provided operating experience with sodium vapor deposits prior to reactor operation. Large component testing in liquid sodium at temperatures up to 1100°F (593°C) has simulated the thermal operating conditions of the FFTF including the evaporation rate of sodium from the surfaces of high temperature sodium. Testing of several plant scale components with various reactor head plug gap configurations has made it possible to evaluate the influence of annuli gap clearance on sodium vapor deposits. The dimensions of the annuli are described, followed by details of the operating conditions of their exposure to sodium vapor and observations of the amount and character of vapor deposits.

- 1547 (HEDL-SA--1518) EXPERIMENTAL FUEL ASSEMBLY IRRADIATION EXPERIENCE IN EBR II. Jackson, R.J.; Kaplan, S.; Schwaile, A. (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1978. Contract EY-76-C-14-2170. 19p. (CONF-790306--8). Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

Irradiation and laboratory tests at EBR-2 of fuel assembly bundles having features prototypic of fast breeder fuel assembly design are described. Much of the work was on fuel bundles design for FFTF.

- 1548 (HEDL-SA--1521) RECENT ADVANCES IN FEATURE EXTRACTION TECHNIQUES BEING DEVELOPED FOR ULTRASONIC EXAMINATION OF AUSTENITIC STAINLESS STEEL WELDS. Mech, S.J.; Emmons, J.S.; Michaels, T.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1978. Contract EY-76-C-14-2170. 10p. (CONF-780829--1). Dep. NTIS, PC A02/MF A01.

From 22. symposium and instrument display; San Diego, CA, USA (28 Aug 1978).

Techniques are described which are being developed to characterize the features found during ultrasonic examination of stainless steel welds which are indicative of defects. Ultrasonic waveforms obtained from both defects and grain boundaries have similar time-domain characteristics. This phenomenon, together with variable signal attenuation and dispersion, is commonly encountered. The problem is to develop feature extraction techniques which will enable the examiner to discriminate reliably between weld defect signals and the other noise. Techniques presented use both time and frequency domain algorithms. The use of these techniques has demonstrated significantly better discrimination than conventional ultrasonic methods.

- 1549 (HEDL-SA--1522) OXYGEN TRANSPORT LIMITATIONS IN USING GETTERS TO CONTROL FUEL CLADDING CHEMICAL INTERACTION. Wilson, C.N. (Hanford Engineering Development Lab., Richland, WA (USA)). 17 Nov 1978. Contract EY-76-C-14-2170. 15p. (CONF-790111--3). Dep. NTIS, PC A02/MF A01.

From International symposium on thermodynamics of nuclear materials; Julich, F.R. Germany (29 Jan 1979).

A series of stainless steel clad $\text{Pu}_{0.25}\text{U}_{0.75}\text{O}/\text{sub 2-x/}$ fuel pins was irradiated in EBR-II to assess placement and oxygen transport limitations using oxygen buffer/getter materials (getters) in LMFBR fuel pins to inhibit fuel-cladding chemical interaction (FCCI). Effective FCCI inhibition required adequate axial dispersal of the getter over the fuel column length. The limiting factor in

getter placement was axial oxygen transport from the fuel to axially-isolated getters. The primary axial oxygen transport mode was concluded to be diffusion of oxygen through the fuel. Gas phase axial oxygen transport is most likely not significant at Δ anti GO_2 values less than the threshold for cladding oxidation.

- 1550 (HEDL-SA--1523-FP) LMFBR PLUTONIUM FUEL DEVELOPMENT AND FABRICATION. Heer, W.F. (Babcock and Wilcox Co., Apollo, PA (USA). Nuclear Materials Div.). 1978. Contract EY-76-C-14 2170. 8p. (CONF-781004--8). Dep. NTIS, PC A02/MF A01.

From Nuclear meeting; Basel, Switzerland (3 Oct 1978).

The history, development, and fabrication of Fast Flux Test Reactor driver fuel pins at the Babcock and Wilcox Company's Parks Township Fabrication Facility in Western Pennsylvania are summarized. The basic process steps are described with emphasis placed on the development efforts which led to the specific production practices used today. Typical conventional production equipment is shown to highlight the standardization which has taken place. Years of continuous production operation have enabled testing and refinement of techniques and equipment systems to the extent that manufacture of LMFBR fuel is now performed routinely within the constraints of all regulatory safety and safeguards criteria, including ALARA (As Low As Reasonably Achievable) control philosophy.

- 1551 (HEDL-SA--1525-FP) DISPLAY OF THE PREDICTIONS OF LARGE FAST REACTOR SAFETY ANALYSIS CODES VIA COMPUTER-GENERATED MOVIES. Lewis, C.H. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1978. Contract EY-76-C-14-2170. 9p. (CONF-780698--1). Dep. NTIS, PC A02/MF A01.

From Northwest computer conference; Victoria, Canada (15 Jun 1978).

A number of computer codes have been written to attempt to predict the consequences of hypothetical accidents. These codes all have one thing in common: they produce reams of paper containing vast quantities of numbers. In an effort to make these numbers more comprehensible, certain graphical techniques have been employed; one of the most successful is the computer-generated movie. The elements necessary for the creation of a plot are presented.

- 1552 (HEDL-SA--1526-FP) EARLY-IN-LIFE THERMAL PERFORMANCE OF UO_2 - PuO_2 FAST REACTOR FUEL. Baker, R.B.; Leggett, R.D. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 11p. (CONF-790306--7). Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

Results from the combined analyses of two thermal performance tests, HEDL P-19 and HEDL P-20 are described. The tests were designed to provide data on the power required to cause incipient fuel melting early in life under conditions prototypic of FFTF driver fuel pins and similar FBR fuel systems.

- 1553 (HEDL-SA--1527-FP) FUEL-CLADDING CHEMICAL INTERACTION IN MIXED-OXIDE FUELS. Lawrence, L.A.; Weber, J.W.; Devary, J.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1978. Contract EY-76-C-14 2170. 14p. (CONF-790306--13).

Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

The character and extent of fuel-cladding chemical interaction (FCCI) was established for UO_2 -25 wt% PuO_2 clad with 20% cold worked Type 316 stainless steel irradiated at high cladding temperatures to peak burnups greater than 8 atom %. The data base consists of 153 data sets from fuel pins irradiated in EBR-II with peak burnups to 9.5 atom %, local cladding inner surface temperatures to 725°C, and exposure times to 415 equivalent full power days. As-fabricated oxygen-to-metal ratios (O/M) ranged from 1.938 to 1.984 with the bulk of the data in the range 1.96 to 1.98. HEDL P-15 pins provided data at low heat rates, approx. 200 W/cm, and P-23 series pins provided data at higher heat rates, approx. 400 W/cm. A design practice for breeder reactors is to consider an initial reduction of 50 microns in cladding thickness to compensate for possible FCCI. This approach was considered to be a conservative approximation in the absence of a comprehensive design correlation for extent of interaction. This work provides to the designer a statistically based correlation for depth of FCCI which reflects the influences of the major fuel and operating parameters on FCCI.

- 1554 (HEDL-SA--1530) OVERVIEW OF U.S. FAST REACTOR IRRADIATION TEST VEHICLES. Claudson, T.T.; Grover, J.M.; McCord, R.B.; Rice, L.H.; Thornton, J.W. (Hanford Engineering Development Lab., Richland, WA (USA)). 25 Apr 1978. Contract EY-76-C-4-2170. 23p. (CONF-7805110--4). Dep. NTIS, PC A02/MF A01.

From Working group on irradiation devices; Petten, Netherlands (18 May 1978).

The United States has conducted extensive irradiation testing on fuels, structural, and absorber materials as a part of the breeder reactor program. Tests have been directed toward basic technology related to fuels and materials, confirmation of material and design selections, and reactor safety analyses. The test reactors used most extensively have been the EBR-II, TReAT, and ETR reactors. The FFTF will be available in the near future and will provide flux levels and test capabilities not previously available in the U.S. The FFTF and EBR-II, sodium-cooled fast reactors rates at 400 and 63 Mwth respectively, are used for steady state irradiations. TReAT and ETR are thermal reactors with provisions for conducting transient tests in special sodium loops. An overview of the design and fabrication of various irradiation test vehicles used in the U.S. Breeder Reactor Program is presented. The test vehicles described have been selected to demonstrate the variety of irradiation tests conducted.

- 1555 (HEDL-SA--1535-FP) HEDL MIXED OXIDE FUEL PIN BREACH EXPERIENCE IN EBR-II. Weber, J.W.; Almassy, M.Y.; Karnesky, R.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1978. Contract EY-76-C-14-2170. 14p. (CONF-790306--12). Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

Mixed plutonium-uranium oxide fuel pins clad with stainless steel are being irradiated to cladding breach in the EBR-II for the Hanford Engineering Development Laboratory's (HEDL) Run-to-Cladding-Breach (RTCB) program. The objectives of these tests are to: (1) establish the burnup capability of mixed oxide fuel pins, (2) identify and characterize

cladding breach mechanisms, (3) determine the effects of cladding breach on neighboring fuel pins and (4) provide data for developing fuel pin failure criteria. Breaches have occurred in 16 HEDL fuel pins in the RTCB program. Examination and characterization has been completed on 13, with the remaining three in some phase of detailed examination. Characterization, includes identification and classification of the probable cause of each breach. The breaches evaluated have been classified as being due to: local cladding overtemperature, subassembly coolant overtemperature, and cladding and wire wrap wear. The cause for one breach has not been conclusively established. The paper reviews characteristics and discusses the probable mechanisms and causes.

1556 (HEDL-SA--1536-FP) MIXED OXIDE RUN-BEYOND-CLADDING-BREACH TESTS IN EBR-II. Washburn, D.F.; Almasy, M.Y.; Langstaff, D.C.; Lambert, J.D.B.; Strain, R.V. (Hanford Engineering Development Lab., Richland, WA (USA); Westinghouse Hanford Co., Richland, WA (USA); Argonne National Lab., IL (USA)). 26 Sep 1978. Contract EY-C-14-2170. 14p. (CONF-790306--9). Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

Fuel pin and subassembly design and quality control during fabrication are intended to ensure that fuel pins can reach goal exposure without cladding breach. However, premature random cladding breaches cannot be precluded. Economic commercial operation of sodium-cooled Fast Breeder Reactors may require operation with breached fuel pins in the core. The alternative to this mode of operation is to remove subassemblies with breached fuel pins at the first indication of cladding breach which is both time consuming and costly. An objective of the U.S. Run-Beyond-Cladding-Breach (RBCB) program is to understand the behavior of mixed oxide fuel during continued irradiation in the post-breach condition. The paper presents the results from the initial RBCB tests of mixed-oxide fuels which were conducted in the Experimental Breeder Reactor-II (EBR-II). Tests conducted to calibrate the EBR-II fission product monitors are also briefly described.

1557 (HEDL-SA--1537-FP) RECORDS MANAGEMENT AT WESTINGHOUSE HANFORD COMPANY. Humason, T.G. (Hanford Engineering Development Lab., Richland, WA (USA)). 23 Aug 1978. Contract EY-76-C-14-2170. 24p. (CONF-7810168--1). Dep. NTIS, PC A02/MF A01.

From Association of records managers and administrators annual conference; Washington, DC, USA (8 Oct 1978).

A program for the acquisition, dissemination, storage, and retrieval of documentation associated with the FFTF project is described.

1558 (HEDL-SA--1540-FP) RADIONUCLIDE TRAP FOR LIQUID METAL COOLED REACTORS. McGuire, J.C.; Brehm, W.F. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1978. Contract EY-76-C-14-2170. 25p. (CONF-781105--80). Dep. NTIS, PC A02/MF A01. From ANS meeting; Washington, DC, USA (12 Nov 1978).

At liquid metal cooled reactor operating temperatures, radioactive corrosion product transport and deposition in the primary system will be sufficiently high to limit access time for maintenance of system components. A radionuclide trap has been developed to aid in

controlling radioactivity transport. This is a device which is located above the reactor core and which acts as a getter, physically immobilizing radioactive corrosion products, particularly ^{54}Mn . Nickel is the getter material used. It is most effective at temperatures above 450°C and effectiveness increases with increasing temperature. Prototype traps have been tested in sodium loops for 40,000 hours at reactor primary temperatures and sodium velocities. Several possible in-reactor trap sites were considered but a location within the top of each driver assembly was chosen as the most convenient and effective. In this position the trap is changed each time fuel is changed.

1559 (HEDL-SA--1543-FP) CESIUM MIGRATION IN LMFBR FUEL PINS. Karnesky, R.A.; Jost, J.W.; Stone, I.Z. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1978. Contract EY-76-C-14-2170. 11p. (CONF-790306--10). Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

The factors affecting the axial migration of cesium in mixed oxide fuel pins and the effects of cesium migration on fuel pin performance are examined. The development and application of a correlated model which will predict the occurrence of cesium migration in a mixed oxide (75 w/o UO_2 + 25 w/o PuO_2) fuel pins over a wide range of fabrication and irradiation conditions are described.

1560 (HEDL-SA--1546) ULTRASONIC TECHNIQUES FOR IN-SERVICE INSPECTION OF AUSTENITIC STAINLESS STEEL WELDS AT 200°C . Mech, S.J.; Emmons, J.S.; Michaels, T.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 37p. (CONF-781008--6). Dep. NTIS, PC A03/MF A01.

From ASNT fall conference; Denver, CO, USA (2 Oct 1978).

Ultrasonic testing (UT) continues to be one of the most viable nondestructive testing (NDT) methods for preservice and inservice inspection of heat transport systems in the nuclear industry. In liquid metal-cooled reactors, these inspections must be conducted at high temperatures (200°C). Consequently, an important objective is to develop methods and equipment to examine austenitic stainless steel weldments at 200°C .

1561 (HEDL-SA--1551-FP) FAST REACTOR FUEL PIN PERFORMANCE REQUIREMENTS FOR OFF-NORMAL EVENTS. Baars, R.E.; Hikido, T.; Hanson, J.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 8p. (CONF-790306--19). Dep. NTIS, MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

Portions of document are illegible.

HEDL is conducting an experimental transient testing program to evaluate the performance of prototypic Fast Flux Test Facility (FFTF) fuel pins up to the cladding integrity limit. The relationship is described of the HEDL/TREAT transient overpower test program to the confirmation of FFTF fuel pin design bases via the FFTF fuel pin design procedure.

1562 (HEDL-SA--1553) FABRICATION AND IRRADIATION PERFORMANCE OF $\text{Pu}_{0.25}\text{U}_{0.75}\text{O}_2$ LOW O/M FUEL. Wilson, C.N. (Hanford Engineering

Development Lab., Richland, WA (USA)). 14 Sep 1978. Contract EY-76-C-14-2170. 18p. (CONF-7810143--3). Dep. NTIS, PC A02/MF A01. From 31. pacific coast meeting of American Ceramic Society; San Diego, CA, USA (25 Oct 1978).

Two very low O/M (1.91) stainless steel clad mixed oxide fuel pins were irradiated to 5 atom % burnup in EBR-II. Low O/M fuel is used to control fuel cladding chemical interaction. Fuel O/M reduction procedures and postirradiation examination data are presented.

1563 (HEDL-SA--1554-FP) B₄C ABSORBER PIN IN-REACTOR RUPTURE TEST. Pitner, A.L.; Hollenberg, G.W.; Jackson, J.L.; Basmajian, J.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1978. Contract EY-76-C-14-2170. 15p. (CONF-781105--89). Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

Portions of document are illegible.

The results of rupture tests on prototypic FFTF B₄C absorber pins are presented. The test in EBR-2 involved an instrumented reference pin and a sealed pin. The observed behavior was compared with calculations by several computer models.

1564 (HEDL-SA--1555-S) ADVANCED ABSORBER ASSEMBLY DESIGN FOR BREEDER REACTORS. SUMMARY. Birney, K.R.; Pitner, A.L.; Hollenberg, G.W. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1978. Contract EY-76-C-14-2170. 6p. (CONF-781105--90). Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

The purpose of this work is to develop an improved breeder reactor absorber assembly design that provides economic and performance benefits relative to the reference FTR absorber assembly design.

1565 (HEDL-SA--1559) MATERIALS COMPATIBILITY IN LIQUID SODIUM. Brehm, W.F. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1978. Contract EY-76-C-14-2170. 38p. (CONF-781071--2). Dep. NTIS, PC A03/MF A01.

From NACE south central conference; Denver, CO, USA (3 Oct 1978).

A review of published literature on sodium technology shows that a great deal is known about the interaction of sodium with the materials and components of liquid-metal-cooled reactors. The transfer of science to technology is rapid. The present generation of sodium-cooled reactors are being built with confidence in the soundness of the sodium systems and a long in-sodium life for components. Materials selection and component design are based on firm, if somewhat conservative, criteria. Continuing work is needed, however, to refine the technological base in such areas as in-reactor measurement of corrosion, effect of prolonged sodium exposure on mechanical properties of metals, and the chemistry of impurities in sodium. This latter topic includes the measurement and control of interstitial elements and radioactive species. Sodium is essentially a benign and easily handled heat transfer medium. The science and technology of liquid metal systems had advanced rapidly. Without a doubt the use of liquid metal heat transfer systems in energy-conversion systems other than nuclear fission reactors will increase as the success of present systems continues to be demonstrated.

1566 (HEDL-SA--1562-FP) THERMOCHEMICAL DATA AND ITS USE IN MODELING CHEMICAL BEHAVIOR IN MIXED-OXIDE FUEL PINS. Gibby, R.L.; Woodley, R.E.; Adamson, M.G.; Johnson, C.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 15p. (CONF-790306--22). Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

The status of US activities to obtain fuel chemistry data is reviewed. Analytical expressions addressing basic needs of all fuel chemistry models are presented. Fission product concentrations during irradiation, oxygen-to-metal (O/M) at beginning-of-life and at burnup, and the potential in fuel-cladding gap at burnup are described. (FS)

1567 (HEDL-SA--1564) IN-CORE GAS RELEASE CAPSULE DEVELOPMENT. Kolowith, R.; Miller, W.C. (Hanford Engineering Development Lab., Richland, WA (USA)). 29 Sep 1978. Contract EY-76-C-14-2170. 28p. (CONF-781105--81). Dep. NTIS, PC A03/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

Portions of document are illegible.

Tests were performed on several candidate fusible plug materials and geometries to demonstrate the feasibility of the fusible plug as a reliable, low-cost gas release mechanism for the Fast Flux Test Facility (FFTF) Fuel Failure Monitoring System In-Core Gas Release Test. The three materials recommended for use in the gas release capsules are eutectic solder, lead and tin. Several plug fabrication methods were investigated. The recommended fabrication method includes using high purity materials, casting the plugs in a nickel tube, electropolishing the nickel tubing and making the plugs in an inert atmosphere glovebox. Sodium exposure tests were conducted and protective coatings were developed.

1568 (HEDL-SA--1564-S) IN-CORE GAS RELEASE CAPSULE DEVELOPMENT. Kolowith, R.; Miller, W.C. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 5p. (CONF-781105--98). Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

Acceptance tests of the FFTF Fuel Failure Monitoring System will use gas releases in a non-critical reactor core to simulate actual fuel failures. The development tests which demonstrate the feasibility of a fusible plug concept as a predictable and reliable in-core gas release mechanism are described.

1569 (HEDL-SA--1574) MECHANICAL PROPERTIES OF TRANSIENT-TESTED IRRADIATED FAST-REACTOR CLADDING. Johnson, G.D.; Hunter, C.W. (Westinghouse Hanford Co., Richland, WA (USA)). 9 Aug 1978. Contract EY-76-C-14-2170. 5p. (CONF-781105--94). Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

Results are presented for three high burnup PNL-11 fuel pins irradiated to goal FFTF exposure levels in testing cladding stress and strain on fast reactor fuel pins subjected to simulated power transient conditions. The mixed oxide fuel with Type 316 SS clad were tested at heating rates of 5.5 k/s and 200 k/s with internal gas pressures of 17.2 to 98.6 MPa. Mechanical property data were obtained. Under short-term transient heating, neutron

volumetric swelling does not change. Cladding strength does not exhibit a continuous degradation with increased neutron fluence, but does show saturation behavior or slight increase at higher fluences. (FS)

- 1570 (HEDL-SA--1575) STRESS ANALYSIS FOR THE DESIGN OF LIQUID METAL PIPING IN THE FAST FLUX TEST FACILITY. Sampson, R.C.; Jagels, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 9p. (CONF-780609--14). Dep. NTIS, MF A01.

From ASME/CSME pressure vessels and piping conference; Montreal, Canada (25 Jun 1978). Portions of document are illegible.

The fast Flux Test Facility (FFTF) is a 400 MWt nuclear facility designed for the exclusive purpose of testing equipment, materials and processes for liquid metal fast breeder reactor power plants. High temperature, low pressure, and severe thermal transient effects are the distinguishing features of the FFTF piping load history. Consequently, a large proportion of the Class 1 piping was designed to comply with ASME Code Case 1331-5 rules for elevated temperature, supplemented by Appendix T of Code Case 1592-3 and RDT-F9-4 for ratchet behavior. This paper outlines the procedures whereby elastic methods for stress analysis were first integrated with the piping design process and then subsequently used to evaluate compliance with Code criteria for extended service at elevated temperature.

- 1571 (HEDL-SA--1576) DESIGN OF THE FMEF AND MATERIAL HANDLING SYSTEMS. Doman, D.R.; Rogers, G.J.; Trego, A.L.; Barker, R.E.; Clark, H.E.; Peschel, W.P. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 36p. (CONF-781105--83). Dep. NTIS, PC A03/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

Final design is over half complete on the Fuels and Materials Examination Facility (FMEF), the hot cell facility for destructive and nondestructive examination of irradiated fuels and materials tested in the Fast Flux Test Facility (FFTF). The overall facility is generally described with detailed information given on systems for transferring fuels and materials for examinations, for solid and liquid radioactive waste, and for equipment maintenance and repair.

- 1572 (HEDL-SA--1579) IN-REACTOR PERFORMANCE OF METHODS TO CONTROL FUEL-CLADDING CHEMICAL INTERACTION. Weber, E.T.; Gibby, R.L.; Wilson, C.N.; Lawrence, L.A.; Adamson, M.G. (Hanford Engineering Development Lab., Richland, WA (USA); General Electric Co., Vallejos, NM (USA)). 1979. Contract EY-76-C-14-2170. 15p. (CONF-790306--15). Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

Inner surface corrosion of austenitic stainless steel cladding by oxygen and reactive fission product elements requires a 50 μ m wastage allowance in current FBR reference oxide fuel pin design. Elimination or reduction of this wastage allowance could result in better reactor efficiency and economics through improvements in fuel pin performance and reliability. Reduction in cladding thickness and replacement of equivalent volume with fuel result in improved breeding capability. Of the factors affecting fuel-cladding chemical interaction (FCCI), oxygen activity within the

fuel pin can be most readily controlled and/or manipulated without degrading fuel pin performance or significantly increasing fuel fabrication costs. There are two major approaches to control oxygen activity within an oxide fuel pin: (1) control of total oxygen inventory and chemical activity (Δ anti GO_2) by use of low oxygen-to-metal ratio (O/M) fuel; and (2) incorporation of a material within the fuel pin to provide in-situ control of oxygen activity (Δ anti GO_2) and fixation of excess oxygen prior to, or in preference to reaction with the cladding. The paper describes irradiation tests which were conducted in EBR-II and GETR incorporating oxygen buffer/getter materials and very low O/M fuel to control oxygen activity in sealed fuel pins.

- 1573 (HEDL-SA--1587) PREHEAT OPERATING EXPERIENCES AT THE FFTF. Tucker, W.R. (Hanford Engineering Development Lab., Richland, WA (USA)). 17 Aug 1978. Contract EY-76-C-14-2170. 22p. (CONF-7809110--2). Dep. NTIS, PC A02/MF A01.

From Plant experience working group meeting; Ibaraki-Ken, Japan (Sep 1978).

The rather extensive test program performed on the FFTF preheat control system resulted in successful sodium fill of one secondary heat transport loop on July 2, 1978. The data obtained during testing and the attendant operating experience gained resulted in some design changes and provided the information necessary to fully characterize system performance. Temperature excursions and deviations from preset limits of only a minor nature were encountered during preheat for sodium fill. The addition of the rate alarm feature was beneficial to operation of the preheat system and allowed early detection and correction of impending excursions.

- 1574 (HEDL-SA--1588) INITIAL ACCEPTANCE TEST EXPERIENCE WITH FFTF PLANT EQUIPMENT. Brown, R.K.; Coleman, K.A.; Mahaffey, M.K.; McCargar, C.G.; Young, M.W. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1978. Contract EY-76-C-14-2170. 35p. (CONF-7809110--3). Dep. NTIS, PC A03/MF A01.

From Plant experience working group meeting; Ibaraki-Ken, Japan (Sep 1978).

The purpose of this paper is to examine the initial acceptance test experience of certain pieces of auxiliary equipment of the Fast Flux Test Facility (FFTF). The scope focuses on the DHX blowers and drive train, inert gas blowers, H and V containment isolation valves, and the Surveillance and In-service Inspection (SISI) transporter and trolley. For each type of equipment, the discussion includes a summary of the design and system function, installation history, preoperational acceptance testing procedures and results, and unusual events and resolutions.

- 1575 (HEDL-SA--1589-6) INTEGRATED LEAK RATE TESTING OF THE FAST FLUX TEST FACILITY REACTOR CONTAINMENT BUILDING. James, E.B.; Farabee, O.A.; Bliss, R.J. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 18p. (CONF-781105--92). Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

Portions of document are illegible. The initial Integrated Leak Rate Test (ILRT) of the Fast Flux Test Facility containment building was performed from May 27 to June 2, 1978. The test was conducted in air with systems vented and with the containment

recirculating coolers in operation. 10 psig and 5 psig tests were run using the absolute pressure test method. The measured leakage rates were .033% Vol/24 hr. and -.0015% Vol/24 hrs. respectively. Subsequent verification tests at both 10 psig and 5 psig proved that the test equipment was operating properly and it was sensitive enough to detect leaks at low pressures. This ILRT was performed at a lower pressure than any previous ILRT on a reactor containment structure in the United States. While the initial design requirements for ice condenser containments called for a part pressure test at 6 psig, the tests were waived due to the apparent statistical problems of data analysis and the repeatability of the data itself at such low pressure. In contrast to this belief, both the 5 and 10 psig ILRT's were performed in a successful manner at FFTF.

- 1576 (HEDL-SA--1593-FP) INSPECTION TECHNIQUES AND PROCESSES FOR CONTROLLING FFTF FUEL QUALITY. Powers, H.G.; Kirchner, T.L.; McLemore, D.R.; Gottschalk, G.P. (Hanford Engineering Development Lab., Richland, WA (USA)). 18 Oct 1978. Contract EY-76-C-14-2170. 20p. (CONF-781105--82). Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

Reliable performance of fuels for the Fast Flux Test Facility (FFTF) is of paramount importance for the FFTF program. Improved inspection techniques and processes are essential for controlling the quality of such fuels. Several considerations necessitated the development and application of improved inspection techniques and processes. These include fuel design, throughput rate, and pre-irradiation characterization requirements. Some of the more significant advances are in the areas of fuel pellet dimensional and weight measurements, fuel pin diameter measurements, and on-line, computerized quality status tracking. Advances in each of these areas are discussed.

- 1577 (HEDL-SA--1598-FP) EFFECTS OF MULTIPLE TRANSIENTS ON FAST REACTOR FUEL PIN CLADDING MECHANICAL PROPERTIES. Duncan, D.R.; Johnson, G.D.; Hunter, C.W. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 7p. (CONF-781105--97). Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

The effects of various TREAT and simulated transient events on subsequent burst strength of 20% cold worked AISI 316 irradiated fast reactor fuel pin cladding were evaluated. Cladding strength after transient exposure was compared to strength prior to transient exposure, using a HEDL correlation of failure temperatures with irradiation and test parameters. The simulated transients and post-transient strength tests were performed in the Fuel Cladding Transient Tester (FCTT), which was developed at HEDL to simulate cladding conditions during in-reactor transient events by pressurizing tubing specimens under transient heating conditions.

- 1578 (HEDL-SA--1604-FP) ALTERNATIVE OXIDE FUEL PELLET FABRICATION FOR IRRADIATION TESTING. Rasmussen, D.E.; Jentzen, W.R.; McCord, R.B. (Hanford Engineering Development Lab., Richland, WA (USA)). 27 Oct 1978. Contract EY-76-C-14-2170. 24p. (CONF-7810143--5). Dep. NTIS, PC A02/MF A01.

From 31. pacific coast meeting of American Ceramic Society; San Diego, CA, USA (25 Oct

1978).

Portions of document are illegible.

Fabrication of experimental breeder reactor fuel pellets by the common cold-press-and-sinter technique for irradiation testing in EBR-II is discussed. Fuel types include mixed thorium-plutonium, UO_2 enriched with 22 weight percent ^{233}U in U , UO_2 enriched with thirty-four weight percent ^{235}U in U , and mixed uranium-plutonium.

- 1579 (HEDL-SA--1605-FP) CLOSED LOOP IN-REACTOR ASSEMBLY (CLIRA): A FAST FLUX TEST FACILITY TEST VEHICLE. Oakley, D.J. (Hanford Engineering Development Lab., Richland, WA (USA)). 12 Sep 1978. Contract EY-76-C-14-2170. 28p. (CONF-781202--42). Dep. NTIS, PC A03/MF A01.

From ASME meeting; San Francisco, CA, USA (10 Dec 1978).

The Closed Loop In-Reactor Assembly (CLIRA) is a test vehicle for in-core material and fuel experiments in the Fast Flux Test Facility (FFTF). The FFTF is a fast flux nuclear test reactor operated for the Department of Energy (DOE) by Westinghouse Hanford Company in Richland, Washington. The CLIRA is a removable/replaceable part of the Closed Loop System (CLS) which is a sodium coolant system providing flow and temperature control independent of the reactor coolant system. The primary purpose of the CLIRA is to provide a test vehicle which will permit testing of nuclear fuels and materials at conditions more severe than exist in the FTR core, and to isolate these materials from the reactor core.

- 1580 (HEDL-SA--1606) FAST FLUX TEST FACILITY ACCEPTANCE TEST PROGRAM AND EARLY RESULTS. Bliss, R.J.; Peckinpaugh, C.L.; Waldo, J.B.; James, E.B. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 31p. (CONF-7809110--1). Dep. NTIS, PC A03/MF A01.

From Plant experience working group meeting; Ibaraki-Ken, Japan (Sep 1978).

Portions of document are illegible.

At the present time, approximately 65% of the Phase 1 and 2 tests have been completed. The major systems which have been tested and are now operational to support sodium fill are summarized. On July 2, 1978 the first secondary loop was filled with sodium and pony motor flow initiated on July 3. Sodium sampling and cold trapping procedures were started immediately thereafter and are proceeding well. The preoperational Integrated Leak Rate Test of the containment building was satisfactorily completed on June 2, 1978. Presently, major emphasis is being placed on completing those test activities which are prerequisites for sodium fill of the remaining two secondary loops and the primary system. In particular, emphasis is being placed on dry refueling system tests and inerted primary cell leak rate testing since these two efforts present the major workloads remaining in the prerequisites to sodium fill of the primary system.

- 1581 (HEDL-SA--1609) FUEL ADJACENCY EFFECTS ON FAST REACTOR CLADDING MECHANICAL PROPERTIES. Hunter, C.W.; Johnson, G.D. (Hanford Engineering Development Lab., Richland, WA (USA)). 6 Dec 1978. Contract EY-76-C-14-2170. 12p. (CONF-790306--21). Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

Simulated transient tests were conducted on 234 cladding specimens from EBR-II irradiated

mixed oxide fuel pins; approximately 75% of the specimens were from the fuel column region, with the remainder from the plenum and below the fuel column. The cladding specimens were taken from the N-E, N-F, PNL-9, PNL-10, PNL-11, P-23A, P-23B, P-23C, and WSA-3 fuel pins irradiated at 15.2 to 37 KW/cm to burnup levels from 11 to 110 MWd/Kg. All the fuel pins used 20% cold worked Type 316 stainless steel cladding. Irradiation temperatures ranged from 370 to 725°C with a peak fluence of 10^{23} n/cm² (E > 0.1 MeV).

- 1582 (HEDL-SA--1625) CONTROL OF RADIONUCLIDE TRANSPORT IN SODIUM-COOLED REACTOR PRIMARY CIRCUITS. Brehm, W.F. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1978. Contract EY-76-C-14-2170. 18p. (CONF-781109--20). Dep. NTIS, PC A02/MF A01.

From Environmental control symposium; Washington, DC, USA (28 Nov 1978).

In order to be economically viable, sodium-cooled reactors must operate at fairly high temperatures (500°C) and with long, uninterrupted fuel cycles. These conditions increase the potential for radioactive corrosion product transport; achieving the goal of long uninterrupted fuel cycles means that release and transport of fission products from breached fuel pins will undoubtedly have to be controlled. This paper shows that control of radioactive material transport is an achievable goal.

- 1583 (HEDL-SA--1630) PMS: WHAT ARE YOU GETTING FOR YOUR MONEY.. Rigney, R.A.; Campbell, R.H. (Hanford Engineering Development Lab., Richland, WA (USA); Department of Energy, Washington, DC (USA). Office of Nuclear Waste Management). 27 Sep 1978. Contract EY-76-C-14-2170. 12p. (CONF-7810120--1). Dep. NTIS, MF A01.

From PMI seminar/symposium; Anaheim, CA, USA (7 Oct 1978).

Portions of document are illegible.

The development and effectiveness of a Performance Measurement System (PMS) for contractors at Hanford Engineering Development Laboratory are discussed. The system was developed in particular for the FFTF project but has been extended to other areas.

- 1584 (HEDL-SA--1631-FP) FUEL CRACK VOLUME PREDICTIONS AND MEASUREMENTS. Fox, G.L.; Randklev, E.H. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1979. Contract EY-76-C-14-2170. 13p. (CONF-790310--5). Dep. NTIS, PC A02/MF A01.

From Specialists workshop on predictive analysis of material dynamics in LMFBR safety expts; Los Alamos, NM, USA (13 Mar 1979).

Void volume within an LMFBR fuel pin has a substantial effect on calculations of cladding strain and lifetime for transient overpower accident analyses. A substantial fraction of this void volume is contained within startup cracks that are still present during fuel irradiation. The paper discusses a crack prediction method and crack measurement techniques, compares predictions with measurements, and includes a sufficient crack fraction data base to be of use to other fuel pin behavior analysts.

- 1585 (HEDL-SA--1634) DESIGN EXPERIENCE: FFTF SHIELDING. Bunch, W.L.; Rathbun, J.L.; Swenson, L.D. (Hanford Engineering Development Lab., Richland, WA (USA); Westinghouse Electric Corp., Madison, PA (USA).

Advanced Reactors Div.). Nov 1978. Contract EY-76-C-14-2170. 17p. (CONF-781117--18). Dep. NTIS, PC A02/MF A01.

From FBK shielding seminar; Obninsk, USSR (13 Nov 1978).

The Fast Flux Test Facility is being built to serve as the primary test facility in the United States for the development of liquid-metal-cooled fast breeder reactors. Shield design philosophy for the reactor was to protect permanent structural members with removable or replaceable components. Significant studies were undertaken to establish the shield design in the reactor support area, to protect the core support structure, and to prevent excessive activation of secondary sodium. Equipment associated with each of the three heat transport system loops are contained in separate shielded cells to permit individual shutdown and isolation maintenance. Design-basis source strengths were calculated and dose rate criteria were established based on anticipated access requirements to provide a basis for the design of plant shields. High density concretes were employed in some walls because of physical constraints established by the limited size of the containment building. Extensive shield acceptance tests are planned to establish the radiation environment throughout the reactor and plant to assure satisfactory performance and for subsequent comparison with design values.

- 1586 (HEDL-SA--1640) LOAD RATING AND DESIGN CRITERIA FOR PIPE SNUBBERS WITH EXTENSION STRUTS. Entz, R.D.; Anderson, M.J. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1979. Contract EY-76-C-14-2170. 7p. (CONF-790615--25). Dep. NTIS, PC A02/MF A01.

From 3. US national congress on pressure vessels and piping; San Francisco, CA, USA (25 Jun 1979).

Snubbing devices are used extensively in nuclear power plant piping to prevent seismic shock forces from causing damaging pipe motions. Most seismic snubbers are designed and load rated by the manufacturer without consideration of attaching structure, unless specifically requested to do so by the purchaser. This paper discusses the additional design constraints imposed by the use of extension struts with the seismic snubbers. The constraints evaluated include load rating reduction as a function of strut length, and strut length limitations to prevent dynamic instability.

- 1587 (HEDL-SA--1641) INELASTIC ANALYSIS OF TWO PIPELINES IN THE FAST FLUX TEST FACILITY. Huang, S.N. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1979. Contract EY-76-C-14-2170. 18p. (CONF-790615--24). Dep. NTIS, PC A02/MF A01.

From 3. US national congress on pressure vessels and piping; San Francisco, CA, USA (25 Jun 1979).

Several complex pipelines of the Fast Flux Test Facility (FFTF) were evaluated using detailed inelastic analysis because they did not satisfactorily comply with the ASME elastic analysis rules. The purpose of the inelastic analysis is to demonstrate that the piping systems, which operate at elevated temperatures, comply with ASME Code requirements under the loading conditions specified in the Design Specification of each pipeline, using inelastic analysis rules. Two pipelines are discussed, the 8-inch Secondary Hot Leg in the Heat Transport System and the 3-inch/4-inch Primary Hot Leg (Ex-module) in the Closed Loop

System. The former is made from type 304SS, operating at 9650F (518.3°C) and designed for a 20 year life; the latter is made from type 316SS, operating at 12000F (648.9°C) and designed for a 10 year life. The MARC general finite element computer program was utilized in the analyses. The pipelines were idealized using the combinations of thin-walled circular closed section beam elements and constant bending 3-node elbow elements. Discussions on the load histograms used, the inelastic strain accumulated, and the effects of creep and fatigue on the lines are given.

- 1588 (HEDL-SA--1642-FP) METHODS OF ASSURING DESIGN ADEQUACY OF SEISMIC III/I SUPPORTS. Summers, G.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1979. Contract EY-76-C-14-2170. 11p. (CONF-790802--78). Dep. NTIS, PC A02/MF A01.

From International conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

After construction of the FFTF plant was virtually complete, special walkdown teams were sent to every cell containing SC I items. These teams, using specially derived zone of influence curves, identified those SC III items which could, if they or their support fail, impact the SC I items. A set of generic criteria was prepared for use by the walkdown teams in the field qualifying the SC III items. These items are referred to as III/I (SC III over SC I). The generic criteria include energy based relationships that can be used to determine whether a SC III item (the missile) has sufficient potential energy to damage a SC I item (the target) even if it does fall on it. Generic criteria were developed to qualify III/I conduct, lights, ducting, piping, pipe supports, and other items. Those items which can't be field qualified with the generic criteria are given detailed analysis. In cases where numerous SC III items threaten a single SC I item or a closely grouped set of SC I items, the SC I item is sometimes protected by providing a barrier between the missile and the target. A description of the analytical basis for the generic criteria is provided in the paper. Examples from FFTF are provided.

- 1589 (HEDL-SA--1643) SHIELDING MATERIALS. Bunch, W.L.; Blackburn, L.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1978. Contract EY-76-C-14-2170. 11p. (CONF-781117--19). Dep. NTIS, PC A02/MF A01.

From FBR shielding seminar; Obninsk, USSR (13 Nov 1978).

The higher neutron flux levels associated with fast reactors, when compared to thermal reactors of the same power, dictate the use of structural steels in the primary reactor shields and the need to design replaceable shield components. The loop-type reactor design being developed in the United States eliminates the need for graphite or other moderating material within the reactor vessel. Sufficient reactor shielding is employed to assure that permanent structural members retain mechanical characteristics comparable to unirradiated material. Removable or replaceable components are designed to retain specific required physical characteristics until replaced. The concretes and compensatory shields used throughout the fast reactor plant are quite similar to those employed in thermal reactor systems. High density concretes containing metallic ores and iron are used where space limitations dictate. Selection of compensatory shield materials is complicated by the diversity of the problems that exist and by the

limited knowledge of the long-term behavior of some materials.

- 1590 (HEDL-SA--1658-S) ANALYSIS OF INTERNAL FUEL MOTION DURING PINEX-2 EXPERIMENT. Padilla, A. Jr.; Baars, R.E.; Porten, D.R.; Randklev, E.H. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1978. Contract EY-76-C-14-2170. 14p. (CONF-790310--6). Dep. NTIS, PC A02/MF A01.

From Specialists workshop on predictive analysis of material dynamics in LMFBR safety expts; Los Alamos, NM, USA (13 Mar 1979).

This paper describes the analyses performed for the PINEX-2 experiment to calculate the ejection of molten fuel into the reflector and fission gas plenum for an internally-vented fuel pin during a simulated 5s/s transient overpower excursion. The LAFM code was used to predict the transient fuel melting and fission gas release, and the HOTPIM and FUMO-I codes were used to predict the fuel ejection. The analytical results were compared with initial data from both the Pinhole-TV Imaging System and the fast-neutron hodoscope, as well as post-transient examinations of the fuel pin.

- 1591 (HEDL-SA--1665) ATTENUATION OF AIRBORNE DEBRIS FROM LMFBR ACCIDENTS. Morewitz, H.A.; Johnson, R.P.; Nelson, C.T.; Vaughan, E.U.; Guderjahn, C.A.; Hilliard, R.K.; McCormack, J.D.; Postma, A.K. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group; Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 15p. (CONF-781022--128). Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

Experimental and theoretical studies have been performed to characterize the behavior of airborne particulates (aerosols) expected to be produced by hypothetical core disassembly accidents (HCDA's) in liquid metal fast breeder reactors (LMFBR's). These aerosol studies include work on aerosol transport in a 20-m high, 850-m³ closed vessel at moderate concentrations; aerosol transport in a small vessel under conditions of high concentration (approx. 1000 g/m³), high turbulence, and high temperature (approx. 2000°C); and aerosol transport through various leak paths. These studies have shown that little, if any, airborne debris from LMFBR HCDA's would reach the atmosphere exterior to an intact reactor containment building.

- 1592 (HEDL-SA--1669) SUMMARY OF HEDL SODIUM FIRE TESTS. Hillard, R.K. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1978. Contract EY-76-C-14-2170. 69p. (CONF-7811137--1). Dep. NTIS, PC A04/MF A01.

From IAEA/INWGER specialists meeting; Cadarache, France (20 Nov 1978).

The sodium fire test program and related studies at the Hanford Engineering Development Laboratory (HEDL) are described. The program is analytical and experimental in scope, with computer code development and experimental verification. Tests have ranged in size from gram quantity laboratory tests to 1600-kg sodium spills. The experimental work is performed in two facilities: the Large Sodium Fire Facility (LSFF) and the Containment Systems Test Facility (CSTF). Sodium fire extinguishment tests which verified the Fast Flux Test Facility (FFTF) secondary sodium fire protection system are described and related information on sodium burning rates and smoke release rates are correlated. The burning rates

are compared to theoretical predictions based on heat and mass transfer analogy, with good agreement. Comparisons with the SOFIRE-II code are also made. Sodium combustion aerosol properties are defined as to chemical and physical nature, settling in closed vessels and effect of added water vapor. The HAA-SB aerosol behavior computer code is compared to tests in the 850 m³ CSTF containment vessel. Sodium spray tests in the CSTF are compared with the SPRAY computer code. An air cleaning program is described, which has the objective of removing high mass concentration sodium combustion aerosols from vented cells and containment buildings. The aerosol mass holding capacity of commercial filters was measured and an aqueous scrubber system is described. The effects of sodium spills on cell structures were investigated, including water release from heated concrete, the reaction of sodium with concrete, the formation and spontaneous recombination of hydrogen, and the ability of steel cell liners to withstand large spills of high temperature sodium without leaking.

- 1593 (HEDL-SA--1671) STATUS OF FFTF STARTUP PROGRAM AND FUTURE FFTF UTILIZATION. Hard, E.N.; Bliss, R.J.; Olson, O.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1978. Contract EY-76-C-14-2170. 52p. (CONF-781014--1). Dep. NTIS, PC A04/MF A01.

From Meeting of Japan section of the American Nuclear Society; Tokyo, Japan (25 Oct 1978).

Portions of document are illegible.

A brief FFTF project description is provided which includes general plant siting information, general layout, plant design parameters, description of principal systems and components, and description of support facilities. The current status of the FFTF project is provided, including status of plant construction, overall status of the plant checkout and test program, status of operating authorization and plant operating procedures and personnel, and status of reactor core components and experiments. Specific information on the acceptance test program and early program results is discussed. The role of FFTF in the future breeder program is described, including its objectives for verification of plant system and components designs and operability and use as an irradiation test facility.

- 1594 (HEDL-SA--1676) PRACTICAL CONSIDERATIONS FROM CONSTRUCTION OF THE FFTF PIPE SUPPORT SYSTEM. Hartsell, J.O. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1979. Contract EY-76-C-14-2170. 30p. (CONF-790488--7). Dep. NTIS, PC A03/MF A01.

From 2. PEWG specialists - meeting; Richland, WA, USA (23 Apr 1979).

Late this year the FFTF is scheduled to initiate criticality and power operation, an event signifying complete resolution of construction problems and compliance with applicable requirements. One of the tasks accomplished to achieve this position was the conduct of cell completion walkthroughs, problem identification, and resolution. Heavily oriented toward the final verification of pipe supports, the cell completion process identified and resolved a series of problems not readily apparent during the detailed design phases of plant construction. It is the objective of this presentation to establish a degree of awareness of these types of problems and of the FFTF techniques employed in resolution. All of the problems depicted herein

have been solved and the installations corrected.

- 1595 (HEDL-SA--1677) ANALYTICAL STUDY OF THE DILATION OF FAST REACTOR FUEL ASSEMBLY DUCTS. Chan, D.P.; Jackson, R.J. (Hanford Engineering Development Lab., Richland, WA (USA)). 15 Nov 1978. Contract EY-76-C-14-2170. 21p. (CONF-790802--75). Dep. NTIS, PC A02/MF A01.

From international conference on structural mechanics in reactor technology; Berlin, F.R. Germany (13 Aug 1979).

An analytic method is presented for determining the dilation of fast reactor fuel assembly ducts. For temperatures where creep is linearly dependent on stress, the method is rigorous in satisfying equilibrium, compatibility and stress-strain equations. Solutions are presented for two cases: (1) a duct with constant pressure differential, (2) a duct with varying pressure differential. Results are in close agreement with finite element results of the MARC-CDC program. The method is used to predict the dilation of the Fast Test Reactor (FTR) ducts under different operating conditions. Presented are the stress, strain and dilation predictions along the duct wall, and the duct dilation variations with its geometric and loading parameters.

- 1596 (HEDL-SA--1678-FP) HOT PRESSING OF EUB. Hollenberg, G.W.; Beutler, P.S. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1979. Contract EY-76-C-14-2170. 26p. (CONF-790420--14). Dep. NTIS, PC A03/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Europium boride pellets with 92% of theoretical density were prepared by hot-pressing. By ball milling the starting powder, hot-pressing temperature could be reduced to 1700°C. The lower hot-pressing temperature provided better control over final density and gave a more desirable microstructure.

- 1597 (HEDL-SA--1679) FFTF IMPLEMENTATION OF TRAINING REQUIREMENTS FOR GOVERNMENT OWNED FACILITIES. Gadeken, A.D.; Izatt, R.D.; Wiborg, J.C. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 26p. (CONF-7804149--1). Dep. NTIS, PC A03/MF A01.

From 3 symposium on the training of nuclear facility personnel; Gatlinburg, TN, USA (29 Apr 1978).

The Fast Flux Facility (FFTF) is a liquid metal cooled, fast flux reactor plant. It is owned by the Department of Energy (DOE) and is operated by the Hanford Engineering Development Laboratory (HEDL) under a DOE contract with Westinghouse Hanford Company. The FFTF is presently undergoing acceptance testing of sodium systems in preparation for initial criticality in August 1979. It will be used to test fuels and materials, to develop associated components and to gain operating experience for future LMFBR's. The FFTF Training Program is a specific example of how training guidelines provided by NRC can be implemented in principle in a DOE owned reactor. The DOE requirements for government owned facilities and the NRC requirements for commercially owned facilities are compared to components of the FFTF Training program. These components will be described, actual status provided and evaluated as to meeting requirements. Cold Plant Qualification (equivalent to NRC Col Plant Licensing) is the next major milestone for FFTF Operator

training. Additional requirements or constraints such as operator time utilization, recruiting and turnover, plant availability and startup testing greatly affect the ability to meet this milestone. These constraints and requirements and the resulting compromises will be evaluated relative to meeting DOE requirements.

- 1598 (HEDL-SA--1683-FP) STEADY STATE IRRADIATION BEHAVIOR OF MIXED OXIDE FUEL PINS IRRADIATED IN EBR-II. Leggett, R.D.; Heck, E.N.; Levine, P.J.; Hilbert, R.F. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1978. Contract EY-76-C-14-2170. 24p. (CONF-790306--11). Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

Results are presented from the steady state fuel irradiation program in EBR-II, which is intended to provide the technology and engineering test data in near-term support of the design and operation of the Fast Flux Test Facility (FFTF) and Fast Breeder Reactors (FBR's) utilizing the reference fuel system - UO_2 - PuO_2 mixed oxide with 25 +/- 5 w/o PuO_2 and 20% CW Type 316 stainless steel cladding. This program is also intended to develop the base for longer term innovative improvements and design changes required for FBR's and develop design criteria and associated computer-based design codes for accurately predicting fuel performance.

- 1599 (HEDL-SA--1686) RECENT AEROSOL TESTS IN THE CONTAINMENT SYSTEMS TEST FACILITY. McCormack, J.D.; Hilliard, R.K.; Postma, A.K. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1978. Contract EY-76-C-14-2170. 21p. Dep. NTIS, PC A02/MF A01.

The Containment Systems Test Facility (CSTF) is a 20-meter high, 850m³ model containment vessel in which various sodium fire aerosol air cleaning tests are studied for purposes of developing and proof testing air cleaning techniques. The first series of tests was designed to study the properties of sodium fire aerosols without any air cleaning and to provide a data base for model verification using large-scale experiments. To date, three such tests (two pool fire and one spray fire) have been completed and are being analyzed. These tests will be discussed, as will plans for the air cleaning tests to follow.

- 1600 (HEDL-SA--1688) W-1 SLSF EXPERIMENT. Simpson, D.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 1 Dec 1978. Contract EY-76-C-14-2170. 33p. (CONF-781144--18). Dep. NTIS, PC A03/MF A01.

From 2. US-Japan seminar on HTGR safety technology; Fuji, Japan (24 Nov 1978).

The Sodium Loop Safety Facility (SLSF) is a doubly-contained, closed sodium loop designed to test full length fast breeder reactor (FBR) fuel pins under simulated hypothetical fast reactor accident conditions. The W-1 experiment has two distinct and separate objectives. The primary objective is to evaluate fuel pin heat release characteristics during loss-of-piping integrity (LOPI) accident flow and power conditions. The Clinch River Breeder Reactor (CRBR) conditions during this accident are being used as representative of FBR's. A progression of four LOPI tests will be conducted to collect data at different fuel pin conditions. These four tests will be performed on fuel pins with (a) fresh, unrestructured

fuel, (b) fresh, restructured fuel, (c) irradiated, uncracked fuel (after steady-state irradiation), and (d) irradiated, cracked fuel (after shutdown and startup). The second objective of the W-1 experiment is to determine stable boiling and recovery limits (boiling window) as a function of fuel pin power.

- 1601 (HEDL-SA--1690-FP) FAST REACTOR NEUTRON ABSORBER MATERIALS. Mahagin, D.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1979. Contract EY-76-C-14-2170. 17p. (CONF-790420--16). Dep. NTIS, PC A02/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

In a fast reactor, nuclear properties limit the number of materials that can be considered for use as neutron absorbers. On a worldwide basis the leading candidates are boron carbide, europium sesquioxide, and europium hexaboride. The performance of these materials is discussed with regard to current and future requirements.

- 1602 (HEDL-SA--1695-FP) PROBABILISTIC METHODS FOR LMFBR APPLICATION. Dutt, D.S.; Chen, K.H.; Stephen, J.D.; O'Reilly, D.B. (Hanford Engineering Development Lab., Richland, WA (USA); General Electric Co., Sunnyvale, CA (USA). Advanced Reactor Systems Dept.; Rockwell International Corp., Palo Alto, CA (USA). Energy Systems Group). 1979. Contract EY-76-C-14-2170. 12p. (CONF-790306--20). Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

The field of statistics and probability, as applied to physical phenomena, is a natural extension of engineering in which reality is modeled. Random variations and their combined effect on system response occur. The paper summarizes the activities at three DOE sponsored laboratories to establish accepted methods to evaluate these random variations, their impact on system performance and to provide assistance in establishing meaningful and quantified product specifications and operating limits. Continued development and refinement of methods to perform probabilistic analysis is required in order for these techniques to become accepted design methods.

- 1603 (HEDL-SA--1703-FP) ANALYSIS OF FFTF RESPONSE TO HYPOTHETICAL EARTHQUAKE WITH POSTULATED PRIMARY PIPING LEAK AND COOLING BY NATURAL CIRCULATION. McCall, T.B.; Franz, G.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1979. Contract EY-76-C-14-2170. 11p. (CONF-790816--66). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The predicted response of the FFTF to a hypothetical earthquake plus a postulated primary pipe leak with subsequent reliance on natural circulation cooling to demonstrate the independence of the primary loops under these conditions is studied. The most pessimistic leak location was selected on the basis of consideration of a sodium spray to maximize the cell gas pressure. The analyses show that the cell gas could not reach the reactor inlet plenum and that the cell pressurization would not interfere with core cooling by the HTS loops. The statistical hot channel coolant temperature would remain below the sodium saturation temperature. Therefore, core coolable geometry would be maintained.

- 1604 (HEDL-SA--1712-FP) HYDROGEN IGNITION AND RECOMBINATION FOLLOWING POSTULATED BREEDER REACTOR ACCIDENTS. Wierman, R.W.; Muhlestein, L.D. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 9p. (CONF-790816--77). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Understanding the inherent response of breeder reactors to hydrogen generated during a major accident is important to reactor safety analyses and in evaluating means of controlling hydrogen to safe levels. Experiments have been conducted at the Hanford Engineering Development Laboratory investigating the ignition, burning, and extinguishment of hydrogen jets under conditions similar to those of postulated reactor accidents. Two extensive series of small-scale tests and two large-scale tests have been performed which indicate the conditions necessary for hydrogen jet ignition and extinguishment. The results and conclusions of the test programs are included.

- 1605 (HEDL-SA--1713-FP) CONTAINMENT AIR CLEANING FOR LMFBRs. Hilliard, R.K.; McCormack, J.D.; Postma, A.K.; Owen, R.K. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 11p. (CONF-790816--56). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

A variety of air cleaning concepts was evaluated for potential use in future sodium-cooled breeder reactors. A 3-stage aqueous scrubber system was selected for large-scale demonstration testing under conditions similar to those postulated for containment venting and purging during reactor melt-through accidents. Two tests were performed in the Containment Systems Test Facility using a quench tank, a jet venturi scrubber and a high efficiency fibrous scrubber in series. The results of two tests with Na_2O_2 and NaOH aerosol and NaI vapor are presented showing >99.9% removal of Na_2O_2 and NaOH and >99.7% for NaI .

- 1606 (HEDL-SA--1714-FP) AEROSOL BEHAVIOR DURING SODIUM SPRAY FIRES AND COMPARISON WITH COMPUTER CODES. McCormack, J.D.; Hilliard, R.K.; Postma, A.K.; Owen, R.K. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1979. Contract EY-76-C-14-2170. 9p. (CONF-790816--79). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The results of sodium spray fires in air in the 850-m³ CSTF vessel are discussed. During these tests, sodium at 600°C was sprayed downward through commercial nozzles located 16 m above the floor. Test AB3 released 48 kg of sodium spray in 140 seconds with 670- μm MMD drop size and produced a maximum pressure increase of 0.041 MPa and 30 g/m³ of airborne sodium compounds. Test NT-1 sprayed 82 kg of sodium at 320 μm MMD drop size over 4.7 hours. In both tests, large stable temperature gradients, not predicted by pre-test SPRAY computer code calculations, were observed. Also, post-test estimations of pressure and aerosol concentration made using SPRAY and HAA-3B codes exceeded actual values measured. Thus, the codes conservatively over-predicted the aerosol leakage potential. The experimental

data and the SPRAY and HAA-3B code calculations are compared and discussed.

- 1607 (HEDL-SA--1716-FP) SODIUM CONCRETE REACTIONS. Colburn, R.P.; Muhlestein, L.D.; Hassberger, J.A.; Mahncke, A.J. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 11p. (CONF-790816--78). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The data from an extensive series of sodium/concrete reaction tests are presented and mechanisms by which the reactions proceed are analyzed. The results indicate water transport and the resulting sodium/water reaction dominate both the chemical energy release and H_2 generation. A mechanism which explains the limited penetration of concrete observed in most of these tests is proposed.

- 1608 (HEDL-SA--1724) QUALITY ASSURANCE FOR BREEDER REACTOR FUEL FABRICATION. Marx, E.R. (Hanford Engineering Development Lab., Richland, WA (USA)). 29 Dec 1978. Contract EY-76-C-14-2170. 24p. (CONF-790420--12). Dep. NTIS, PC A02/MF A01.

From International symposium on ceramics in nuclear waste management; Cincinnati, OH, USA (30 Apr 1979).

Fuel performance in the Fast Flux Test Facility (FFTF) depends on fabrication of fuel to rigorous quality standards. The quality program including Management, Procurement, Fabrication, Inspection, Records, and Audits is discussed as well as unique mixed oxide fuel inspections such as homogeneity inspection, analytical chemistry, and nondestructive fissile assay.

- 1609 (HEDL-SA--1725-FP) COMPARISON OF A FISSION-GAS EFFECTS IN A TRANSIENT OVERPOWER TEST (HUT 5-7A) TO FRAS3 CODE PREDICTIONS. Gruber, E.E.; Randklev, E.H. (Argonne National Lab., IL (USA); Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 9p. (CONF-790816--67). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Fission gas has an important bearing on fuel dynamics during reactor transients. Fission-gas bubble sizes and densities, both within grains and on grain boundaries, are characterized as functions of radial location at the axial midplane in studies of PNL-9 fuel microstructures before and after the HUT 5-7A (PNL 9-25) TREAT test. The FRAS3 code, being developed to model fission-gas effects in reactor transients, is applied to analyze the results of the HUT 5-7A test are presented to illustrate the observed phenomena and the validity of the modeling approach.

- 1610 (HEDL-SA--1727-FP) REMOTE FABRICATION OF PELLET FUELS FOR UNITED STATES BREEDER REACTORS. Dahl, R.E.; Nyman, D.H.; Yatabe, J.M. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1979. Contract EY-76-C-14-2170. 12p. (CONF-790602--90). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

Goal of the program is to demonstrate the feasibility of fabricating breeder fuel in a remotely operated and maintained mode by 1985. Development for pellet fuel fabrication is in

the engineering stage with much of the equipment for ceramic unit operations in final design or currently under testing. Results to date confirm that remote fabrication of pellet fuels is feasible. Several of the processes and equipment items are described in this report.

- 1611 (HEDL-SA--1728-FP) NEUTRON SOURCE STRENGTH IN FFTF FUEL AS A FUNCTION OF IRRADIATION. Ombrellaro, P.A.; Petrowicz, N.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1979. Contract EY-76-C-14-2170. 20p. (CONF-790602--74). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

The neutron source strength associated with fuel irradiated in the FFTF reactor is important in determining the reactivity of the core when it is shut down or being refueled. The monitoring of subcritical reactivity for FFTF will be done with the Modified Source Multiplication (MSM) technique which requires an accurate knowledge of the neutron source strength in high exposure fuel elements replaced with fresh fuel elements in a shut down reactor. As a means of obtaining the most accurate source evaluations possible with existing nuclear data for FFTF irradiated fuel, a technique for generating one-group actinide cross sections for use in isotopic buildup and depletion codes was developed. A further objective of the work was to use the cross sections and recently evaluated spontaneous fission and (α, n) reaction neutron yield data to calculate the neutron source strength in typical compositions of FFTF-type fuel as a function of flux level and irradiation time and then to compare the results with those of a previous study to assess the differences in the nuclear data. This paper will highlight the important aspects of this study and present some results.

- 1612 (HEDL-SA--1729-FP) NUCLEAR STARTUP, TESTING AND CORE MANAGEMENT OF THE FTR. Bennett, R.A.; Daughtry, J.V.; Harris, R.A.; Jones, D.H.; Nelson, J.W.; Rawlins, J.A.; Rothrock, R.B.; Sevenich, R.A.; Zimmerman, B.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1979. Contract EY-76-C-14-2170. 18p. (CONF-790933--3). Dep. NTIS, PC A02/MF A01.

From IAEA international symposium on fast reactor physics; Aix-en-Provence, France (24 Sep 1979).

The Fast Test Reactor (FTR) is a sodium-cooled, mixed-oxide-fueled, 400 MW(Th) fast reactor designed for irradiation testing of FB fuels and materials. The reactor is located near Richland, Washington, and is operated by the Westinghouse Hanford Company for the US Department of Energy. The FTR is presently undergoing nonnuclear startup tests, in preparation for the initial fuel loading. The reactor is provided with special irradiation facilities to enhance its experimental capabilities.

- 1613 (HEDL-SA--1731-FP) MEASUREMENT OF CLADDING STRAIN DURING SIMULATED TRANSIENT TESTS. Duncan, D.R.; Johnson, G.D.; Hunter, C.W.; Hanson, J.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 20 Jul 1979. Contract EY-76-C-14-2170. 13p. (CONF-790816--73). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

A diametral extensometer was developed and employed during temperature ramp tests with the

Fuel Cladding Transient Tester (FCTT). Plastic strain measurements were performed using unirradiated 20X cold-worked AISI 316 stainless steel tubing ramped at 5.6 and 111°C/s with internal pressures from 3.4 to 93.1 MPa. Results demonstrated that plastic deformation can occur at stresses well below the conventional 0.2% yield strength and that most deformation in such tests occurs in the final 50°C before failure. Postirradiation tests were performed on fuel pin cladding irradiated to 5.8×10^{22} n/cm ($E > 0.1$ MeV) with irradiation temperatures to 540°C. The tests showed that, for test pressures of 17.2 MPa or less, the stress-strain behavior was unchanged from unirradiated material behavior although the strains at failure were greatly decreased.

- 1614 (HEDL-SA--1735-FP) SAFETY ANALYSIS OF EXPERIMENTS PROPOSED FOR IRRADIATION IN FFTF. Stepniewski, D.D.; Van Keuren, J.C. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1979. Contract EY-76-C-14-2170. 8p. (CONF-790816--71). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

A safety analysis of all experiments to be irradiated in FFTF is required to demonstrate that the experiment does not constitute a hazard to the reactor or the public. This paper summarizes the review procedure, the scope of the analysis and safety acceptance criteria.

- 1615 (HEDL-SA--1736-FP) FFTF CONTAINMENT OF HYPOTHETICAL ACCIDENTS. Peak, R.D.; Martin, H.C.; Jensen, R.L.; Stepniewski, D.D.; Hale, J.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1979. Contract EY-76-C-14-2170. 18p. (CONF-790816--70). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The FFTF facility was evaluated for the consequences of an HCDA followed by failure of in-vessel post-accident heat removal, reactor vessel melt-through, and release of core debris and sodium coolant to the reactor cavity. Two cases are presented based on parameters considered to represent upper limits for rates of chemical and thermal attack of the reactor cavity concrete containment structure. The reactor containment building temperature, pressure, and leak rate histories were computed with the CACECO code which provided input into the HAA-3C code for prediction of aerosol behavior, and to the COMRADEX-H code for prediction of radioactivity dispersion. The resultant 30-day doses at the site boundary were judged to be acceptable considering the conservatism in the analysis and the low probability of the event.

- 1616 (HEDL-SA--1737-FP) PINEX-2 EXPERIMENT: CONCEPT VERIFICATION OF AN INHERENT SHUTDOWN MECHANISM FOR HCDA'S. Porten, D.R.; Padilla, A. Jr.; Baars, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1979. Contract EY-76-C-14-2170. 13p. (CONF-790816--68). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

PINEX-2 was a dual-objective experiment to continue development of the Pinhole Imaging (PINEX) system for monitoring fuel motion, and to further demonstrate the feasibility of internal fuel motion as a mechanism for mitigating the consequences of hypothetical

LMFBR accidents. A special fuel pin with a pre-fabricated central hole in the fuel, insulator, and reflector was pre-irradiated in the General Electric Test Reactor and subjected to a simulated 5\$/sec transient overpower excursion in the Transient Reactor Test (TREAT) facility. Approximately 10 grams of molten fuel were ejected upward from the fuel region into the reflector and fission gas plenum.

- 1617 (HEDL-SA--1741-FP) APPLICATION OF OPTIMAL ESTIMATION TECHNIQUES TO FFTF DECAY HEAT REMOVAL ANALYSIS. Nutt, W.T.; Additon, S.L.; Parziale, E.A. (Hanford Engineering Development Lab., Richland, WA (USA)). 20 Jul 1979. Contract EY-76-C-14-2170. 11p. (CONF-790816--69). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The verification and adjustment of plant models for decay heat removal analysis using a mix of engineering judgment and formal techniques from control theory are discussed. The formal techniques facilitate dealing with typical test data which are noisy, redundant and do not measure all of the plant model state variables directly. Two pretest examples are presented.

- 1618 (HEDL-SA--1743-FP) RESPONSE OF FFTF CORE TO PROTECTED REACTIVITY ADDITION TRANSIENTS. Yee, A.K.; Baars, R.E.; Stepnewski, D.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1979. Contract EY-76-C-14-2170. 16p. (CONF-790816--62). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The response of the FFTF core to protected reactivity insertion events was evaluated. Reactivity addition transients ranging from .05 cents/s to 3\$/s have been considered. The evaluation method is based on a calculational model which predicts cladding strain from modified fuel-cladding differential thermal expansion. The results show that for all ramp rates considered, the Plant Protection System (PPS) controls consequences to required limits. Comparisons made between predicted fuel damage and results of TREAT transient tests support the conservatism of the results.

- 1619 (HEDL-SA--1744-FP) FFTF REGULATORY REVIEW FOR OPERATING AUTHORIZATION. Schade, A.R.; Simpson, D.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 7p. (CONF-790816--63). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The Project has undertaken design studies of concepts capable of providing added containment safety margins. Since the events, for which these features provide consequence reductions, are beyond the design basis accident classification, the conceptual designs would not be designed as Engineered Safety Features. A final decision has not been made by the Department of Energy as to the disposition of this advice. It is the Project's plan to develop sufficient design details for these features that construction schedules and cost estimates can be formulated. This information supplemented by the continuing analytical and experimental program to assess the adequacy of FFTF containment margins will enable a decision to be made as to additional features to be added to FFTF and their schedule for

implementation.

- 1620 (HEDL-SA--1747-FD) FISSION GAS BEHAVIOR IN MIXED-OXIDE FUEL DURING TRANSIENT OVERPOWER. Randklev, E.H.; Treibs, H.A.; Mastel, B.; Baldwin, D.L. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 13p. (CONF-790816--86). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Fission gas behavior can be important in determining fuel pin and core performance during a reactor transient. The results are presented of examinations characterizing the changes in microstructural distribution and retention of fission gas in fuel for a series of transient overpower (50 cents/s) tested mixed-oxide fuel pins and their steady state siblings.

- 1621 (HEDL-SA--1749-FP) EFFECTS OF STOICHIOMETRY ON CLADDING REACTION IN MIXED-OXIDE FUEL. Lawrence, L.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1979. Contract EY-76-C-14-2170. 21p. (CONF-790602--75). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

Initial fuel stoichiometry (oxygen-to-metal ratio, O/M) has been established as one of the major parameters influencing the character and extent of fuel-cladding chemical interaction (FCCI) in UD_2 --25 wt% PuO_2 fuels clad with 20X cold worked type 316 stainless steel. How the character of the interaction changes with initial O/M and the O/M dependence that has been established for the depth of interaction are discussed. The data base for this analysis consists of 134 data sets from P-23 series fuel pins irradiated in EBR-II. As-fabricated O/M's ranged from 1.938 to 1.984 with the bulk of the data in the range 1.96 to 1.98. Fuel pins operated to 9.5 atom %, with local cladding inner surface temperatures to 725°C, for exposure times to 415 equivalent full power days at heat rates of approx. 400 W/cm.

- 1622 (HEDL-SA--1750) FATIGUE-CRACK PROPAGATION RESPONSE OF TWO NICKEL-BASE ALLOYS IN A LIQUID SODIUM ENVIRONMENT. Mills, W.J.; James, L.A. (Hanford Engineering Development Lab., Richland, WA (USA)). 23 Jan 1979. Contract EY-76-C-14-2170. 31p. (CONF-790615--20). Dep. NTIS, PC A03/MF A01.

From 3. US national congress on pressure vessels and piping; San Francisco, CA, USA (25 Jun 1979).

The elevated temperature fatigue-crack propagation response of Inconel 600 and Inconel 718 was characterized within a linear-elastic fracture mechanics framework in air and low-oxygen liquid sodium environments. The crack growth rates of both nickel-base alloys tested in liquid sodium were found to be considerably lower than those obtained in air. This enhanced fatigue resistance in sodium was attributed to the very low oxygen content in the inert sodium environment. Electron fractographic examination of the Inconel 600 and Inconel 718 fatigue fracture surfaces revealed that operative crack growth mechanisms were dependent on the prevailing stress intensity level. Under low growth rate conditions, Inconel 600 and Inconel 718 fracture surfaces exhibited a faceted, crystallographic morphology in both air and sodium environments. In the higher growth rate regime, fatigue striations were observed;

however, striations formed in sodium were rather ill-defined. These indistinct striations were attributed to the absence of oxygen in the liquid sodium environment. Striation spacing measurements were found to be in excellent agreement with macroscopic growth rates in both environments.

- 1623 (HEDL-SA--1751) SEISMIC DESIGN AND DEVELOPMENT FOR FAST REACTORS: A DESIGN-APPLICATION PERSPECTIVE WITH DIRECTIONS FOR IMPROVEMENT. Severud, L.K. (Hanford Engineering Development Lab., Richland, WA (USA)). 19 Jul 1979. Contract EY-76-C-14-2170. 29p. (CONF-790816--64). Dep. NTIS, PC A93/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Applications of seismic design criteria and qualification methods to the US breeder reactor projects have developed new findings, improvements in design methods, and identified areas for further development. Discussions are presented regarding site free field motion, soil-structure interaction, equipment response spectra, piping, snubbers and support design analyses, dynamic decoupling, seismic qualification testing, and protection of Seismic Category I components from Non-Category I equipment failures.

- 1624 (HEDL-SA--1752-FP) SURVIVAL OF PERIPHERAL PINS DURING A TOP-HCDA. Yung, S.C.; Wilburn, N.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1979. Contract EY-76-C-14-2170. 19p. (CONF-790816--65). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

This study addresses the survival of peripheral pins within an LMFBR subassembly during an unprotected transient overpower (TOP) Hypothetical Core Disruptive Accident (HCDA) by consideration of intrasubassembly incoherencies. A continuous analysis was made vs time from the initiation of the accident up to the point where the power decreases to a quasi-steady state for a Beginning-of-Cycle-4 (BOC-4) core of Fast Test Reactor (FTR), 0.5\$/sec ramp case. Blockage was assumed to be formed after fuel pin's failure and effects due to blockages were examined. The study concludes that most peripheral pins within an LMFBR subassembly indeed will survive.

- 1625 (HEDL-SA--1753-S(Rev.)) OBSERVATIONS OF IN-REACTOR ENDURANCE AND RUPTURE LIFE FOR FUELED AND UNFUELED FTR CLADDING. Lovell, A.J.; Christensen, B.Y.; Chin, B.A. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 21p. (CONF-790602--81(Rev.)). 02.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

Reactor component endurance limits are important to nuclear experimenters and operators. This paper investigates endurance limits of 316 CW fuel pin cladding. The objective of this paper is to compare and analyze two different sets of FTR fuel pin cladding data. The first data set is from unfueled pressurized cladding irradiated in the Experimental Breeder Reactor No. II (EBR-II). This data set was generated in an assembly in which the temperature was monitored and controlled. The second data set contains observations of breached and unbreached EBR-II test fuel pins covering a large range of temperature, power and burnup conditions.

- 1626 (HEDL-SA--1754-FP) OBSERVATIONS OF IN-REACTOR STRAIN FOR FUELED AND UNFUELED FTR CLADDING. Gilbert, E.R.; Makenas, B.J.; Wilson, D.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1979. Contract EY-76-C-14-2170. 15p. (CONF-790602--83). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

It has been demonstrated that equations derived from in-reactor creep and swelling in unfueled pressurized tubes of 20% CW AISI 316 stainless steel can be used to predict strains in prototypic FTR mixed-oxide (UO_2 -- PuO_2) fuel pins. For fast neutron fluences below $6 \times 10^{22}n/cm^2$ the strains were small (less than one percent) and good agreement was found (within 0.1 percent diametral strain) between the equations and the fuel pin strains. This paper describes an extension of the earlier study to fast neutron fluences up to $11 \times 10^{22}n/cm^2$.

- 1627 (HEDL-SA--1755-FP) COMPARISON OF IN-REACTOR AND OUT-OF-REACTOR FUEL PIN CLADDING STRAIN UNDER TRANSIENT LOADING. Duncan, D.R.; Hunter, C.W.; Thielges, J.R.; Hikido, T.; Hanson, J.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 9p. (CONF-790602--76). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

Results of two recent fuel pin transient tests performed in Mark II-C sodium loops in the TREAT reactor are compared with out-of-reactor transient test results. The TREAT pins were irradiated in EBR-II in the WSA-3 experiment; steady-state irradiation and TREAT test parameters are presented. Out-of-reactor transient tests in the Fuel Cladding Transient Tester (FCTT) measured the deformation and failure behavior of unirradiated specimens and irradiated fuel pin cladding specimens under constant internal gas pressure loading and transient temperature conditions. All fuel pins were 20% cold-worked AISI 316 stainless steel.

- 1628 (HEDL-SA--1756-5) LONG-TERM IN-REACTOR CORROSION OF STAINLESS STEEL CLADDING IN LIQUID SODIUM. Jensen, B.W.; Weber, J.W.; Jost, J.W. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 6p. (CONF-790602--82(Summ.)). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

In-reactor sodium corrosion of 20% cold-worked 316 stainless steel cladding was previously reported on HEDL P-23 series mixed-oxide fuel pins exposed in EBR-II for up to 4608 hours. This report extends those studies to 11,424 hours. Data from three HEDL subassemblies, P23A, P23B, and P23C, are included in this study. Nominal sodium velocities were 1.95 m/sec, 1.83 m/sec, and 2.18 m/sec, respectively. These 5.84 mm diameter fuel pins contained 25% PuO_2 - 75% UO_2 mixed-oxide fuel operating at a peak power of 440 W/cm. Time-averaged cladding outer surface temperature ranged from 400 to 712°C. Axial temperature gradients were approximately 960°C/m for the P23A and P23B subassemblies and 755°C/m for the P23C subassembly. Oxygen concentration in the sodium was approximately 1 ppm.

- 1629 (HEDL-SA--1759-FP) EFFECT OF A TIME VARYING POWER LEVEL IN EBR-II ON MIXED-OXIDE FUEL BURNUP. Stone, I.Z.; Jost, J.W.; Baker,

- R.B. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 7p. (CONF-790602--80). Dep. NTIS, PC A02/MF A01.
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
A refined prediction of burnup of mixed-oxide fuel in EBR-2 is compared with measured data. The calculation utilizes a time-varying power factor and results in a general improvement to previous calculations.
- 1630 (HEDL-SA--1760-FP) INFLUENCE OF OXYGEN-METAL RATIO ON MIXED-OXIDE FUEL PERFORMANCE. Lawrence, L.A.; Leggett, R.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1979. Contract EY-76-C-14-2170. 21p. (CONF-790602--79). Dep. NTIS, PC A02/MF A01.
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
The fuel oxygen-to-metal ratio (O/M) is recognized as an important consideration for performance of uranium-plutonium oxide fuels. An overview of the effects of differing O/M's on the irradiation performance of reference design mixed-oxide fuel in the areas of chemical and mechanical behavior, thermal performance, and fission gas behavior is presented. The pellet fuel has a nominal composition of 75 wt% UO_2 + 25 wt% PuO_2 at a pellet density of approx. 90% TD. for nominal conditions this results in a smeared density of approx. 85%. The cladding in all cases is 20% CW type 316 stainless steel with an outer diameter of 5.84 to 6.35 mm. O/M has been found to significantly influence fuel pin chemistry, mainly FCCI and fission product and fuel migration. It has little effect on thermal performance and overall mechanical behavior or fission gas release. The effects of O/M (ranging from 1.938 to 1.984) in the areas of fuel pin chemistry, to date, have not resulted in any reduction in fuel pin performance capability to goal burnups of approx. 8 atom% or more.
- 1631 (HEDL-SA--1766) STRESS ANALYSIS OF CONICAL FLUED HEADS FOR FFTF LIQUID METAL PIPING ANCHORS. Sampson, R.C. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1979. Contract EY-76-C-14-2170. 33p. (CONF-790615--17). Dep. NTIS, PC A03/MF A01.
From 3. US national congress on pressure vessels and piping; San Francisco, CA, USA (25 Jun 1979).
Flued heads for liquid metal piping were designed to function as structural anchors for liquid metal piping and as atmospheric seals between compartments. Their design was complicated by the need to provide high strength and rigidity as piping anchors, yet to have the flexibility needed to accommodate severe thermal differential temperature gradients under both steady state and transient conditions. Conical flued head design configurations in sizes from 1" to 28" were evolved by the repeated application of detailed thermal and stress analysis methods. Stress indices characterizing the response to each basic load type, were derived from the finite element analyses, followed by evaluations of performance by the elevated temperature ASME Code rules for elastic analysis and by the simplified one-dimensional analysis under the rules for inelastic analyses. Based upon this work, slight modifications of the design were made before authorizing production in order to enhance the resistance to creep damage. It was later demonstrated that the most critically stressed flued heads under the most extreme operating conditions complied with the Code rules for detailed inelastic stress analysis.
- 1632 (HEDL-SA--1769) ANALYSIS OF PIPING SYSTEMS WITH NONLINEAR SUPPORTS SUBJECTED TO SEISMIC LOADING. Barta, D.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1979. Contract EY-76-C-14-2170. 23p. (CONF-790615--18). Dep. NTIS, PC A02/MF A01.
From 3. US national congress on pressure vessels and piping; San Francisco, CA, USA (25 Jun 1979).
An analytical study of effects of nonlinearities in piping supports on response to seismic excitation is presented. Response calculations for simplified single degree of freedom nonlinear models are used to illustrate sensitivity to stiffness variations, lost motion and impact damping. Seismic responses of typical spans of various sizes of piping supported by both linear and nonlinear constraints are compared to assess the support load magnifications due to impacting. The idealized nonlinear piping support models are integrated with a finite element model of a large piping system. Time domain seismic responses of the nonlinear piping system are compared to loads determined by a standard linearized seismic response spectra technique.
- 1633 (HEDL-SA--1770) RADIOLOGICAL ANALYSIS OF HYPOTHETICAL ACCIDENTS BY COMPUTER. Martin, H.C.; Hale, J.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1979. Contract EY-76-C-14-2170. 14p. (CONF-790816--74). Dep. NTIS, PC A02/MF A01.
From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).
The radiological analyses of extreme hypothetical accidents were performed almost wholly by computer techniques. Major analytical codes used were RIBD, CACECO, SPRAY, HAA, and COMRADEX. This paper describes the analyses, the modeling techniques, and transition programs that edited data output from one code into a form suitable for input to another. Although not sacrificing a conservative stance, the degree of realism afforded by these analyses shows that the projected consequences could be acceptable. Furthermore the techniques which have been developed are applicable to most LMFBR accident analyses.
- 1634 (HEDL-SA--1771) DESIGN ANALYSIS OF LIQUID METAL PIPE SUPPORTS. Margolin, L.L.; LaSalle, F.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1979. Contract EY-76-C-14-2170. 15p. (CONF-790615--19). Dep. NTIS, PC A02/MF A01.
From 3. US national congress on pressure vessels and piping; San Francisco, CA, USA (25 Jun 1979).
Design guidelines pertinent to liquid metal pipe supports are presented. The numerous complex conditions affecting the support stiffness and strength are addressed in detail. Topics covered include modeling of supports for natural frequency and stiffness calculations, support hardware components, formulas for deflection due to torsion, plate bending, and out-of-plane flexibility. A sample analysis and a discussion on stress analysis of supports are included. Also presented are recommendations for design improvements for increasing the stiffness of pipe supports and which were utilized in the FFTF system.
- 1635 (HEDL-SA--1772) INSTRUMENT CALIBRATION

PROGRAM FOR STARTUP ON FFTF. Swanson, C.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1979. Contract EY-76-C-14-2170. 27p. (CONF-790807--2). Dep. NTIS, PC A03/MF A01.

From Topical conference on reactor operating experience; Arlington, TX, USA (5 Aug 1979).

Calibration of plant instruments and control equipment is a key plant startup activity. Clearly defined calibration requirements are needed to effectively guide this activity. A master-loop/sub-loop concept developed for use at the FFTF to help solve this problem is described. Startup testing methods related to the calibration activity are also described and critiqued.

1636 (HEDL-SA--1773) EXPERIENCE WITH SIMPLIFIED INELASTIC ANALYSIS OF FFTF TEST ASSEMBLIES. Winkel, B.V. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1979. Contract EY-76-C-14-2170. 18p. (CONF-790615--22). Dep. NTIS, PC A02/MF A01.

From 3. US national congress on pressure vessels and piping; San Francisco, CA, USA (25 Jun 1979).

Included in the more recent design activities for the Fast Flux Test Facility (FFTF) are a series of test assemblies which will contain experiments in the FFTF reactor. Due to high operating temperatures and severe thermal transients, inelastic analysis is generally required to demonstrate structural adequacy. Based upon previous FFTF experience, a number of simplifications have been made to reduce the inelastic analysis effort. Significant time and cost savings have resulted.

1637 (HEDL-SA--1787-FP) FAILURE LOCATION ANALYSIS FOR TAGGED REACTOR ASSEMBLIES. Figg, J.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1979. Contract EY-76-C-14-2170. 12p. (CONF-790698--2). Dep. NTIS, PC A02/MF A01.

From Society of Women Engineers national student conference; San Francisco, CA, USA (27 Jun 1979).

The location of defective LMFBR fuel pins by the determination of gas tag isotopic ratios is discussed. The application of this method to the FFTF Reactor briefly described.

1638 (HEDL-SA--1794) EXPERIENCE IN THE USE OF FBR CORE COMPONENT STRUCTURAL DESIGN CRITERIA AS APPLIED TO FFTF. Hecht, S.L. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 10p. (CONF-790615--23). Dep. NTIS, PC A02/MF A01.

From 3. US national congress on pressure vessels and piping; San Francisco, CA, USA (25 Jun 1979).

User gained experience resulting from trial applications of proposed structural design guidelines for Fast Breeder Reactor (FBR) core components is presented. This work was done supporting the design analyses process for consumable core components for the Fast Test Reactor (FTR) of the Fast Flux Test Facility (FFTF). The proposed guidelines were found to be more comprehensive and generally easier to apply than those methods previously used. Component evaluation required a minimum amount of detailed inelastic analysis, primarily through the use of simplified inelastic analysis methods, as given in the guidelines. A major shortcoming of this draft criteria/guidelines is a lack of supporting irradiated material properties. Some areas of guidance

given seems ambiguous and may be non-conservative, particularly those related to stress classification unique to FBR environments. Further verification of these areas appears to be in order.

1639 (HEDL-SA--1795) FFTF STATUS AND TEST RESULTS. Noordhoff, B.H. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1979. Contract EY-76-C-14-2170. 219p. (CONF-7904117--1). Dep. NTIS, PC A10/MF A01.

From American Nuclear Society meeting; Los Angeles, CA, USA (19 Apr 1979).

Portions of document illegible.

A general description of the FFTF is presented concerning the construction progress; status of startup and selected startup test results; and planned experiment program during and subsequent to the startup testing program. The FFTF is a 400 MWt, sodium cooled test reactor plant consisting of a fast-neutron reactor fueled by mixed plutonium oxide-uranium oxide fuel elements, three independent heat transport loops, extensive instrumentation, fuel handling equipment, and auxiliary systems which support the main reactor and cooling systems. The FFTF contains no steam generators, electrical generating equipment, or breeding blankets. Heat generated in the reactor is released to the ambient atmosphere by twelve large sodium-to-air heat exchangers.

1640 (HEDL-SA--1828) TITANIUM OXIDE CESIUM GETTERS FOR LOW O/M FBR FUEL PINS. Wilson, C.N.; Gibby, R.L.; Weber, E.T. (Hanford Engineering Development Lab., Richland, WA (USA)). 15 Aug 1979. Contract EY-76-C-14-21790. 15p. (CONF-791045--7). Dep. NTIS, PC A02/MF A01.

From American Ceramic Society conference; New Orleans, LA, USA (14 Oct 1979).

Fission product cesium may contribute to cladding strain in low oxygen-to-metal ratio (O/M) FBR fuel pins through localized reaction with fuel or UO_2 blanket pellets. Titanium oxide pellets were laboratory irradiation tested as candidate cesium getters for FBR fuel pins. Results indicate satisfactory performance.

1641 (HEDL-SA--1835) CONFIDENCE LEVEL IN THE CALCULATIONS OF HCDA CONSEQUENCES USING LARGE CODES. Nguyen, D.H.; Wilburn, N.P. (Hanford Engineering Development Lab., Richland, WA (USA)). 20 Apr 1979. Contract EY-76-C-14-2170. 7p. (CONF-791103--61). Dep. NTIS, PC A02/MF A01.

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

The probabilistic approach to nuclear reactor safety is playing an increasingly significant role. For the liquid-metal fast breeder reactor (LMFBR) in particular, the ultimate application of this approach could be to determine the probability of achieving the goal of a specific line-of-assurance (LOA). Meanwhile a more pressing problem is one of quantifying the uncertainty in a calculated consequence for hypothetical core disruptive accident (HCDA) using large codes. Such uncertainty arises from imperfect modeling of phenomenology and/or from inaccuracy in input data. A method is presented to determine the confidence level in consequences calculated by a large computer code due to the known uncertainties in input invariables. A particular application was made to the initial time of pin failure in a transient overpower HCDA calculated by the code MELT-IIIA in order to demonstrate the method. A probability

distribution function (pdf) for the time of failure was first constructed, then the confidence level for predicting this failure parameter within a desired range was determined.

- 1642 (HEDL-SA--1838-FP) CORA: TRANSIENT ANALYSIS CODE FOR A CLUSTER OF REACTOR CORE ASSEMBLIES. Johnson, H.G. (Hanford Engineering Development Lab., Richland, WA (USA)). 25 Apr 1979. Contract EY-76-C-14-2170. 29p. (CONF-790602--78). Dep. NTIS, PC A03/MF A01.

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

The CORA code is a steady state/transient, core thermal hydraulics code for the FFTF Reactor. A brief overview of the code development and use is presented.

- 1643 (HEDL-SA--1839) THERMAL BOWING OF WIRE-WRAPPED FUEL PINS IN FAST REACTORS. Chan, D.P.; Jackson, R.J. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 24p. (CONF-791103--68). Dep. NTIS, PC A02/MF A01.

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

The mechanism of thermal bowing of fuel pins is discussed. Finite element models for thermal bowing analysis using MARC-CDC and BEAMCREEP programs are presented. Analytical results are in reasonably close agreement with each other. Thermal bowings of edge fuel pins in various FTR core positions are predicted. Conservative predictions indicate that the reference edge pins with 30.48 cm (12 in.) axial pitch may touch the duct wall in as little as 1000 h or irradiation. The results further indicate that the axial stress induced in the cladding by thermal bowing is 4 times as much as that due to fission gas release. However, the combined axial stress level is still below that to cause stress rupture in the design life of the pin.

- 1644 (HEDL-SA--1846) FFTF MATERIALS OF CONSTRUCTION AND SURVEILLANCE PLANS. Shober, F.R. (Hanford Engineering Development Lab., Richland, WA (USA)). 4 May 1979. Contract EY-76-C-14-2170. 35p. Dep. NTIS, PC A03/MF A01.

Portions of document are illegible.

FFTF REACTOR; REACTOR MATERIALS; REACTOR COMPONENTS; CONSTRUCTION; SPECIFICATIONS; INSPECTION; MATERIALS TESTING

- 1645 (HEDL-SA--1850) DESIGN AND CONSTRUCTION OF THE FUELS AND MATERIALS EXAMINATION FACILITY. Burgess, C.A. (Hanford Engineering Development Lab., Richland, WA (USA)). 15 Nov 1979. Contract EY-76-C-14-2170. 32p. (CONF-791103--59). Dep. NTIS, PC A03/MF A01.

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

Final design is more than 85 percent complete on the Fuels and Materials Examination Facility, the facility for post-irradiation examination of the fuels and materials tests irradiated in the FFTF and for fuel process development, experimental test pin fabrication and supporting storage, assay, and analytical chemistry functions. The overall facility is generally described with specific information given on some of the design features. Construction has been initiated and more than 10% of the construction contracts have been awarded on a fixed price basis.

- 1646 (HEDL-SA--1854FP) INTEGRATED QUALITY STATUS AND INVENTORY TRACKING SYSTEM FOR FFTF DRIVER FUEL PINS. Gottschalk, G.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1979. Contract EY-76-C-14-2170. 14p. (CONF-791117--18). Dep. NTIS, PC A02/MF A01.

From ANS topical conference; Kiawah Island, SC, USA (26 Nov 1979).

An integrated system for quality status and inventory tracking of Fast Flux Test Facility (FFTF) driver fuel pins has been developed. Automated fuel pin identification systems, a distributed computer network, and a data base are used to implement the tracking system.

- 1647 (HEDL-SA--1860) CREEP RELAXATION OF FUEL PIN BENDING AND OVALING STRESSES. Chan, D.P.; Jackson, R.J. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1979. Contract EY-76-C-14-2170. 27p. (CONF-791205--16). Dep. NTIS, PC A03/MF A01.

From ASME winter annual meeting; New York, NY, USA (2 Dec 1979).

Analytical methods for calculating fuel pin cladding bending and ovaling stresses due to pin bundle-duct mechanical interaction taking into account nonlinear creep are presented. Calculated results are in close agreement with finite element results by MARC-CDC program. The methods are used to investigate the effect of creep on the FTR fuel cladding bending and ovaling stresses. It is concluded that the cladding of 316 SS 20% CW and reference design has high creep rates in the FTR core region to keep the bending and ovaling stresses to low levels.

- 1648 (HEDL-SA--1868) EFFECTS OF TRANSIENT POWER HISTORIES ON THE INTRASUBASSEMBLY FAILURE INCOHERENCIES. Yung, S.C. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1979. Contract EY-76-C-14-2170. 51p. (CONF-791103--64). Dep. NTIS, PC A04/MF A01.

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

The effects of different transient power histories on the intrasubassembly failure incoherencies in an unprotected Transient Overpower (TOP) Hypothetical Core Disruptive Accident (HCDA) are analyzed. The computational tool and other relevant points of present analysis are described. The computational results from the analysis are also discussed. The summary of the present study is stated.

- 1649 (HEDL-SA--1881-FP) DEVELOPMENT OF AN INSTRUMENTED MATERIALS IRRADIATION TEST FOR FFTF. Grover, J.M.; Kaulitz, D.C.; Krogness, J.C. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 4p. (CONF-791103--63). Dep. NTIS, PC A02/MF A01.

From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).

The development of an advanced test assembly for use in the Fast Flux Test Facility is described. The facility will be used to characterize alloys for in reactor use. (FS)

- 1650 (HEDL-SA--1895-FP) FABRICATION OF C.5-INCH DIAMETER FBR MIXED OXIDE FUEL PELLETS. Rasmussen, D.E.; Benecke, M.W.; McCord, R.B. (Hanford Engineering Development Lab., Richland, WA (USA)). 12 Sep 1979. Contract EY-76-C-14-2170. 43p. (CONF-791045--4). Dep. NTIS, PC A03/MF A01.

From American Ceramic Society conference; New Orleans, LA, USA (14 Oct 1979).

Large diameter (0.535 inch) mixed oxide fuel pellets for Fast Breeder Reactor application were successfully fabricated by the cold-press-and-sinter technique. Enriched UO_2 , $PuO_2=UO_2$, and PuO_2-ThO_2 compositions were fabricated into nominally 90% theoretical density pellets for the UO_2 and $PuO_2=UO_2$ compositions, and 88% and 93% T.D. for the PuO_2-ThO_2 compositions. Some processing adjustments were required to achieve satisfactory pellet quality and density. Furnace heating rate was reduced from 200 to 50°C/h for the organic binder burnout cycle for the large, 0.535-inch diameter pellets to eliminate pellet cracking during sintering. Additional preslugging steps and die wall lubrication during pressing were used to eliminate pressing cracks in the PuO_2-ThO_2 pellets.

- 1651 (HEDL-SA--1903-FP) HIGH VOLTAGE INSTRUMENT CABLES FOR 650°C IN-VESEL BREEDER REACTOR SERVICE. Cannon, C.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1979. Contract EY-76-C-14-2170. 16p. (CONF-791037--5). Dep. NTIS, PC A02/MF A01.

From IEEE nuclear science symposium; San Francisco, CA, USA (17 Oct 1979).

MgO insulated nuclear instrument cables have been developed for use in the Fast Flux Test Facility (FFTF) and breeder reactor in-vessel environments. Prototype cables have been successfully fabricated and tested: voltage Breakdown Pulse Noise (BPN) thresholds in excess of 800 Vdc and cable resistivities in excess of $10^8 \Omega \cdot ft$ were achieved at temperatures to 650°C. Factors demonstrated to strongly affect breakdown phenomena and insulation resistance in bulk cable were insulation and backfill gas purity (especially as affected by fabrication processing), compaction density, and high temperature annealing. Substitution of copper electrical conductors did not adversely affect cable performance if fabrication parameters were appropriately adjusted. Preliminary gamma irradiation tests on bulk MgO cable indicate that the gamma flux decreases insulation resistance by an order of magnitude to $5 \times 10^9 \Omega \cdot ft$ at 450°C.

- 1652 (HEDL-SA--1919-FP) FFTF PRIMARY SYSTEM TRANSITION TO NATURAL CIRCULATION FROM LOW REACTOR POWER. Bouchey, G.D.; Additon, S.L.; Nutt, W.T. (Department of Energy, Richland, WA (USA). Richland Operations Office; Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1980. Contract EY-76-C-14-2170. 18p. (CONF-800226--4). Dep. NTIS, PC A02/MF A01.

From Specialists meeting on decay heat removal and natural circulation in FBR's; Upton, NY, USA (28 Feb 1980).

Plans for reactor and primary loop natural circulation testing in the Fast Flux Test Facility (FFTF) are summarized. Detailed pretest planning with an emphasis on understanding the implications of process noise and model uncertainties for model verification and test acceptance are discussed for a transition to natural circulation in the reactor core and primary heat transport loops from initial conditions of 5% of rated reactor power and 75% of full flow.

- 1653 (HEDL-SA--1920S) COMPUTER SIMULATION OF NATURAL CIRCULATION IN FFTF SECONDARY LOOPS. Beaver, T.R.; Turner, D.M.; Additon, S.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1979. Contract EY-76-C-14-2170. 10p. (CONF-800226--2). Dep.

NTIS, PC A02/MF A01.

From Specialists meeting on decay heat removal and natural circulation in FBR's; Upton, NY, USA (28 Feb 1980).

A thermal/hydraulic model of the FFTF secondary heat transport loop has been calibrated against transient natural circulation test data collected March to May 1979. The tests verified that the transition to natural convective flow could be effected from near isothermal conditions without excessive cooling at the air dump heat exchangers. Key empirical parameters of pressure drop and heat loss were found to be at 88% and 81% of pretest estimates, respectively. Pretest piping thermal transport and flow calculational models required no further revision to produce good agreement with test data.

- 1654 (HEDL-SA--1921-FP) POST TEST EVALUATION OF NATURAL CIRCULATION IN FFTF SECONDARY LOOPS. Beaver, T.R.; Turner, D.M.; Additon, S.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1980. Contract EY-76-C-14-2170. 19p. (CONF-800226--5). Dep. NTIS, PC A02/MF A01.

From Specialists meeting on decay heat removal and natural circulation in FBR's; Upton, NY, USA (28 Feb 1980).

Two transient tests in one of the FFTF secondary heat transport loops were performed (March to May 1979) in order to verify that the transition to natural convective flow could be effected from near isothermal refueling conditions without excessive cooling at the air dump heat exchangers. Following the tests, the best estimate computer model was calibrated against the data, yielding in the process insights about the loop parameters and the probable suitability of the model structure. Key empirical parameters of pressure drop and heat loss were found to be at 62% and 81% of the pretest safety evaluation model parameter values, respectively. Pretest piping thermal transport and flow calculational models required no further revision to produce good agreement with the test data. Additional detail in the air dump heat exchanger heat loss model, accounting for long structural thermal time constants, was found to be necessary to obtain model agreement with transient outlet temperature data. The pretest model had conservatively employed steady state heat losses for transient calculations.

- 1655 (HEDL-SA--1931) DISSOLUTION OF FFTF VENDOR FUEL. Lerch, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1979. Contract EY-76-C-14-2170. 26p. (CONF-791086--2). Dep. NTIS, PC A03/MF A01.

From US/United Kingdom information exchange on dissolution of nuclear fuel; Windscale, UK (16 Oct 1979).

Dissolution experiments were performed on FFTF vendor fuel (both mechanically mixed and coprecipitated) during 1974, 1975, and 1976. A marked improvement was noted in the completeness of fuel dissolution from 1974 to 1976. The reason for this is unknown but may have been attributable to slight changes in fuel fabrication conditions. In general, the bulk of the fuel pellets tested dissolved to greater than 99.9% in nitric acid alone.

- 1656 (HEDL-SA--1932) EQUIPMENT FOR INSPECTION OF AUSTENITIC STAINLESS STEEL PIPE WELDS. Boehmer, W.D.; Horn, J.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 20 Aug 1979. Contract EY-76-C-14-2170. 4p. (CONF-791008--4). Dep. NTIS, PC

A02/MF A01.

From ASNT fall conference; St Louis, MO, USA (15 Oct 1979).

A computer controlled ultrasonic scanning system and a data acquisition and analysis system have been developed to perform the in-service inspection of welds in stainless steel sodium piping in the Fast Flux Test Facility. The scanning equipment consists of a six axis motion mechanism and control system which allows full articulation of an ultrasonic transducer as it follows the circumferential pipe welds. The data acquisition and analysis system consists of high speed ultrasonic waveform digitizing equipment, dedicated processors to perform on-line analysis, and data storage and display equipment.

- 1657 (HEDL-SA--1935) DISSOLUTION OF MIXED OXIDE FUEL AS A FUNCTION OF FABRICATION VARIABLES. Lerch, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1979. Contract EY-76-C-14-2170. 54p. (CONF-791086--3). Dep. NTIS, PC A04/MF A01.

From US/United Kingdom information exchange on dissolution of nuclear fuel; Windscale, UK (16 Oct 1979).

Dissolution properties of mechanically blended mixed oxide fuel were very dependent on the six fuel fabrication variables studied. Fuel sintering temperature, source of PuO_2 and PuO_2 content of the fuel had major effects: (1) as the sintering temperature was increased from 1400 to 1700°C, pellet dissolution was more complete; (2) pellets made from burned metal derived PuO_2 were more completely dissolved than pellets made from calcined nitrate derived PuO_2 which in turn were more completely dissolved than pellets made from calcined nitrate derived PuO_2 ; (3) as the PuO_2 content decreased from 25 to 15 wt % PuO_2 , pellet dissolution was more complete. Preferential dissolution of uranium occurred in all the mechanically blended mixed oxide. Unirradiated mixed oxide fuel pellets made by the Sol Gel process were generally quite soluble in nitric acid. Unirradiated mixed oxide fuel pellets made by the coprecipitation process dissolved completely and rapidly in nitric acid. Fuel made by the coprecipitation process was more completely dissolved than fuel made by the Sol Gel process which, in turn, was more completely dissolved than fuel made by mechanically blending UO_2 and PuO_2 as shown below. Addition of uncomplexed fluoride to nitric acid during fuel dissolution generally rendered all fuel samples completely dissolvable. In boiling 12M nitric acid, 95 to 99% of the plutonium which was going to dissolve did so in the first hour. Irradiated mechanically blended mixed oxide fuel with known fuel fabrication conditions was also subjected to fuel dissolution tests. While irradiation was shown to increase completeness of plutonium dissolution, poor dissolubility due to adverse fabrication conditions (e.g., low sintering temperature) remained after irradiation.

- 1658 (HEDL-SA--1945) NATURAL CIRCULATION IN FFTF. Addison, S.L.; Bouchey, G.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1979. Contract EY-76-C-14-2170. 40p. (CONF-791074--1). Dep. NTIS, PC A03/MF A01.

From US/USSR exchange meeting on fast reactor safety; Moscow, USSR (8 Oct 1979).

This paper provides a review of the evaluation of natural circulation in the Fast Flux Test Facility (FFTF), a loop type reactor plant. The emphasis is on the role of natural circulation in decay heat removal; however, free convection during either operation at

power or normal shutdown does influence some aspects of the design and these are mentioned. In treating decay heat removal the topics discussed include steady state loop performance and transient dynamics for conditions immediately after scram and for the longer term which involves different considerations. The review summarizes complex dynamics, specific to the FFTF design evaluation, which particularly illustrate large available margins for decay heat removal (possibly involving coolant boiling). Also addressed are modelling considerations and the status of experimental programs. To date, design adequacy has been assessed largely by a combination of component tests and system analyses, with design margins increased to accommodate uncertainties. Presently planned tests, when completed, will provide an empirical base for characterizing system dynamic behavior in the range most relevant to current LMFBR designs and will demonstrate FFTF decay heat removal capability directly.

- 1659 (HEDL-SA--1948) RADIOACTIVE CORROSION PRODUCT RELEASE POTENTIAL IN SODIUM-COOLED REACTORS. Colburn, R.P.; Taylor, G.R. (Hanford Engineering Development Lab., Richland, WA (USA); Westinghouse Electric Corp., Pittsburgh, PA (USA). Advanced Reactors Div.). Sep 1979. Contract EY-76-C-14-2170. 23p. Dep. NTIS, PC A02/MF A01.

A mathematical analysis of the rate of release and deposition of radioactive corrosion products from the core of liquid sodium cooled reactors is outlined and numerical results presented as a function of reactor design parameters for a number of reactors. It is shown that the activity release potential of advanced sodium-cooled reactors (FFTF and CRBRP) is about two orders of magnitude larger than that of older reactors such as EBR-II, and RAPSODIE. Radiation levels up to 6 R/hr are predicted for FFTF and CRBRP primary system components after 20 years of operation. The model can be used to predict the sensitivity of activated corrosion product release to changes in operating parameters for various existing and planned reactors.

- 1660 (HEDL-SA--1950=FP) CARBON IN SODIUM: A STATUS REVIEW OF THE USA R AND D WORK. McCown, J.J.; Bagnall, C. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170. 42p. (CONF-791149--1). Dep. NTIS, PC A03/MF A01.

From IWGFR specialists meeting on carbon in sodium; Harwell, UK (26 Nov 1979).

A review is presented concerning R and D work on carbon in sodium with reference to LMFBR primary coolant circuits. The chemistry of carbon in sodium, analysis and monitoring of carbon in sodium, carbon meters, and problems of carbon in sodium are described. 31 references. (FS)

- 1661 (HEDL-SA--1958) FFTF SAFETY REVIEW. Simpson, D.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1979. Contract EY-76-C-14-2170. 6p. Dep. NTIS, PC A02/MF A01.

Key features of the FFTF design related to safety are described. The independent safety review process is also discussed.

- 1662 (HEDL-SA--1959) FFTF FUEL STORY. Claudson, T.T.; Bardsley, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 1979. Contract EY-76-C-14-2170.

9p. Dep. NTIS, PC A02/MF A01.

The development and irradiation testing of FFTF fuel is reviewed.

- 1663 (HEDL-SA--1973-FP) FFTF PURIFICATION SYSTEM PERFORMANCE DURING FILL AND INITIAL TEMPERATURE ASCENSION. Bloom, G.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1979. Contract EY-76-C-14-2170. 24p. Dep. NTIS, PC A02/MF A01.

Design parameters of the sodium purification systems on the Fast Flux Test Facility are given. Sodium and argon cover gas impurity changes observed during initial sodium fill and initial reactor heatup to 800°F are discussed. Sodium Purification System operation during these critical sodium impurity control periods is described.

- 1664 (HEDL-SA--1983, pp 20-40) EBR-II AND FFTF IRRADIATION EXPERIMENTS. Gilbert, E.R.; Paxton, M.M. 1979.

From Seminar on fast reactor cladding materials; Richland, WA, USA (12 Nov 1979).

Structural materials experiments were conducted in EBR-II using several different types of capsules. Subcapsule containing specimens were immersed in static sodium. The subcapsules are insulated from the outer capsule by a gas annulus which determines the irradiation temperature. A second type of noninstrumented capsule utilizes heat pipes for high temperature tests (650 to 750°C) where temperature fluctuations must be minimized. These heat pipes provide compensatory control to minimize effects of power fluctuation on temperature. In addition, instrumented vehicles using thermocouples and temperature controllers have been used for high temperature creep tests requiring accurate temperature measurement. These experimental methods were used to investigate the effects of temperature, temperature change, stress, and fluence dependence of in-reactor creep. Special tests have also yielded data on the effects of stress on swelling.

- 1665 (HEDL-SA--1983, pp 62-76) MICROSTRUCTURAL OBSERVATIONS OF AISI 316 STAINLESS STEEL FROM HIGH BURNUP FUEL PINS. Brager, H.R.; Berting, F.M.; Garner, F.A. 1979.

From Seminar on fast reactor cladding materials; Richland, WA, USA (12 Nov 1979).

Microstructural data were obtained from three fuel pins clad with 20% cold worked drawn tubing made from 316 stainless steel. The cladding was irradiated at a temperature in the range 370 to 600°C to a peak neutron fluence of 10×10^{22} n/cm² ($E > 0.1$ MeV) in the EBR-II. Additionally, cladding from a fuel pin utilizing a steel identical to that used to clad FFTF first core fuel pins was examined after irradiation at comparable temperatures to a peak neutron fluence of 6×10^{22} n/cm² ($E > 0.1$ MeV). The microstructure from the mid-wall regions of the cladding specimens has been compared with that from specimens made from the same tubing but irradiated in non-fueled experiments. The comparison shows that the microstructural development is not significantly influenced by the presence of the fuel but rather is dependent on irradiation temperature and neutron fluence. The microstructure of the fuel pin cladding shows that void nucleation and growth and that Frank loop nucleation are temperature dependent. The microstructures support the concept of an interaction model for unfaulting of Frank loops, exhibit a temperature-independent saturation density of dislocation networks, and

illustrate that void nucleation occurs heterogeneously. It would appear that nucleation of voids and swelling rate increases are associated with the emergence of nickel-rich precipitate phases including γ' (Ni₃Si). Type 316 stainless steel under irradiation tends to evolve towards a final saturation microstructure in which dislocation and Frank loop concentrations are independent of the starting microstructure.

- 1666 (HEDL-SA--1983, pp 169-186) CREEP BEHAVIOR OF OX16H15M3B STEEL AND COMPARISON WITH AISI 316. Gilbert, E.R.; Chin, B.A. (Hanford Engineering Development Lab., Richland, WA). 1979.

From Seminar on fast reactor cladding materials; Richland, WA, USA (12 Nov 1979).

Pressurized tube creep specimens were manufactured from OX16H15M3B tubing supplied by the USSR. These specimens are currently being irradiated in Experimental Breeder Reactor II (EBR-II). Four specimens (0, 33, 66 and 132 MPa) are being irradiated at 425°C in Capsule B-217b of the AA-IIa experiment while five specimens (0, 7.5, 18.9, 32 and 64 MPa) are being irradiated at 650°C in Capsule B-279a of the AA-IIc experiment. The 650°C specimens were discharged from the reactor after EBR-II Run 99 at a fast fluence ($E > 0.1$ MeV) of 4.2×10^{22} n/cm². These specimens were measured and re-encapsulated for reirradiation beginning with Run 10 (May 1979). The 425°C specimens were discharged after EBR-II Run 100 at a fast fluence ($E > 0.1$ MeV) of 2.8×10^{22} n/cm². These specimens were measured and re-encapsulated for reirradiation which began in Run 102 (July 1979). This paper reviews the results of the measurements and compares them with creep data generated in the BOR-60 fast reactor on hexagonal channels and US design equations developed from results on AISI 316 pressurized tube specimens.

- 1667 (HEDL-SA--2014) ULTRASONIC OFF-NORMAL IMAGING TECHNIQUES FOR UNDER SODIUM VIEWING. Michaels, T.E.; Horn, J.E. (Hanford Engineering Development Lab., Richland, WA (USA)). 6 Dec 1979. Contract EY-76-C-14-2170. 18p. (CONF-800231--1). Dep. NTIS, PC A02/MF A01.

From 3. international conference on nondestructive evaluation in the nuclear industry; Salt Lake City, UT, USA (11 Feb 1980).

Advanced imaging methods have been evaluated for the purpose of constructing images of objects from ultrasonic data. Feasibility of imaging surfaces which are off-normal to the sound beam has been established. Laboratory results are presented which show a complete image of a typical core component. Using the previous system developed for under sodium viewing (USV), only normal surfaces of this object could be imaged. Using advanced methods, surfaces up to 60 degrees off-normal have been imaged. Details of equipment and procedures used for this image construction are described. Additional work on high temperature transducers, electronics, and signal analysis is required in order to adapt the off-normal viewing process described here to an eventual USV application.

- 1668 (HEDL-SA--2022) DATA SYSTEMS IN FFTF AND EBR-II. Warrick, R.P.; Ritter, W.M. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1980. Contract EY-76-C-14-2170. 25p. (CONF-800366--2). NTIS, PC A02/MF A01.

From Specialists' meeting on LMFBR

instrumentation and control; Tokyo, Japan (31 Mar 1980).

This paper describes the Data System used to monitor operation and collect experimental data in FFTF. This data system has evolved since initial inception from a relatively simple, single computer system monitoring a relatively few (approx. 1000) instrument channels important for operation to one which has increased capability to support the long-range testing needs in FFTF. The system, while still relatively simple, now contains multiple computers which normally perform independent functions. The computers, however, provide backup processing for certain simple tasks. Operator interfacing is provided through CRT's. The output capabilities of the system are described. A description of the Data System in EBR-II is also included.

- 1669 (HEDL-SA--2023) NUCLEAR INSTRUMENTATION SYSTEM DESIGN IN FFTF AND CRBRP. Warrick, R.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1980. Contract EY-76-C-14-2170. 30p. (CONF-800366--1). NTIS, PC A03/MF A01.

From Specialists' meeting on LMFBR instrumentation and control; Tokyo, Japan (31 Mar 1980).

The Nuclear Instrumentation System installed in the Fast Flux Test Facility (FFTF) is described. The Nuclear Instrumentation System includes equipment for monitoring neutron flux levels from shutdown to full power. Detector location and mounting provisions are described. The design basis for equipment design is provided. Detailed discussion of startup testing in FFTF follows a brief discussion of pre-delivery development work and testing. Finally, a description of the Nuclear Instrumentation System planned for the Clinch River Breeder Reactor Plant is provided.

- 1670 (HEDL-SA--2066-FP) THERMOELASTIC BUCKLING OF PLATES IN A CYLINDRICAL GEOMETRY AGAINST AN ELASTIC BACK SUPPORT. Simmons, L.D.; Wierman, R.W. (Hanford Engineering Development Lab., Richland, WA (USA)). 1980. Contract EY-76-C-14-2170. 16p. (CONF-800360--1). NTIS, PC A02/MF A01.

From International conference on thermal stresses in materials and structures in severe thermal environments; Blacksburg, VA, USA (19 Mar 1980).

A plate which is fixed at its edges to a strong edge support structure will develop large compressive stresses when heated from ambient temperature more rapidly than the support structure. Determining the response of the plate to this situation requires stability analysis to ascertain whether the plate might buckle, or whether the constrained thermal expansion will lead to compressive stresses exceeding the yield point because it did not buckle. A special case is considered here, both analytically and experimentally, in which the plate is curved slightly into a cylindrical shape and the convex face of the plate is against a supporting surface. This case is more complex because the buckling mode will be a harmonic rather than the fundamental mode which is usually encountered.

- 1671 (HEDL-TC--517) SUMMARY STRESS RESULTS FOR THE HEAT TRANSPORT SYSTEM PIPING AND COMPONENTS. Bartholf, L.W.; Edwards, P.A.; Ketchum, H.B. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). May 1976. Contract EY-76-C-14-2170. 375p. Dep. NTIS, PC A16/MF A01.

Portions of document are illegible.

FFTF REACTOR; REACTOR COOLING SYSTEMS; PIPES; HEAT EXCHANGERS; STRESS ANALYSIS; PUMPS; VALVES

- 1672 (HEDL-TC--572 (Rev.1)) ANALYSIS OF FFTF REACTOR CAVITY LINERS UNDER EX-VESSEL PAHR CONDITIONS. Polentz, L.M. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1976. Contract EY-76-C-14-2170. 175p. AT.

Portions of document are illegible.

A detailed analysis is presented of the effects of a hot-sodium spill caused by a hypothetical reactor-vessel melt-through on the steel liner plates in the FFTF reactor cavity. The effects of steam pressurization, thermal shock, and thermal stress are analyzed. In order to evaluate the effects of the steam pressure, a steam venting concept is included in the analysis. Complete flow and stress analyses are given including the effects of biaxial strain. It is concluded that the steel liner panels would yield plastically a small amount, but that there would be no danger of gross failure provided that the steam could escape from calculated margin of safety is greater than 1.6.

- 1673 (HEDL-TC--807) CRITICALITY ANALYSIS FOR FFTF FUEL ASSEMBLIES AND FUEL PINS. Morford, R.J.; Marr, D.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1977. Contract EY-76-C-14-2170. 37p. Dep. NTIS, PC A03/MFA01.

A series of calculations is presented assessing the criticality aspects of specific configurations of FTR fuel. Results for regularly spaced arrangements of fuel pins or fuel assemblies as a function of spacing in each of four media (water, Mobiltherm, sodium, or air) are given. This information is intended to be of a broad general nature conservatively applicable both to normal fuel configurations likely to be encountered in the operation of the FFTF and to accident condition configurations postulated for criticality review purposes.

- 1674 (HEDL-TC--902-1) HEDL CONTRIBUTION TO SRL FUEL RECYCLE PROGRAM. QUARTERLY REPORT, JANUARY--MARCH 1977. Fletcher, J.F. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1977. Contract EY-76-C-14-2170. 44p. Dep. NTIS, PC A03/MF A01.

Research on LWR fuel cycle is being done in the following categories: economic studies (sensitivity analysis of LMFBR fuel fabrication costs), spent fuel receipt and storage (failure of PWR and BWR fuel assemblies), fuel materials preparation or finishing processes, reduction of TRU waste generation, and environmental impacts. 12 tables. (DLC)

- 1675 (HEDL-TC--902-2) HEDL CONTRIBUTION TO SRL FUEL RECYCLE PROGRAM. QUARTERLY REPORT FOR APRIL, MAY, AND JUNE 1977. Fletcher, J.F. (comp.). (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1977. Contract EY-76-C-14-2170. 42p. Dep. NTIS, PC A03/MF A01.

Work is being done in five LWR categories: economic studies (capital cost of 1 MT/day LMFBR fuel fabrication facility); spent fuel receipt and storage (fuel failure, radioactivity releases); fuel materials preparation (review of safeguards and standards); reduce generation of TRU waste (treatment of HEPA filters); and environmental effects (hydrologic data for WTRAN code and modifications to DOSE code). (DLC)

- 1676 (HEDL-TI--1022) NATURAL CIRCULATION HEAT TRANSFER TESTING WITH A SIMULATED FULL-SCALE LMFB 217-PIN ELECTRICALLY HEATED FUEL ASSEMBLY: TEST DATA. Millburg, M.L.; Hassberger, J.A.; Boasso, C.J. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1978. Contract EY-76-C-14-2170. 240p. AT.
- The heat transfer characteristics of a Clinch River Breeder Reactor Plant (CRBRP) nearly prototypic fuel assembly in a liquid sodium environment were investigated under low flow conditions indicative of natural circulation. The facilities, test plan and tests themselves were described in detail and discussed in an earlier report, "Natural Circulation Heat Transfer Testing with a Simulated Full-Scale LMFB 217-Pin Electrically Heated Fuel Assembly", HEDL-TME 77-3, June 1977. The report presented provides the data obtained during the test activities. These data were developed to supplement analytical predictions of assembly heat transfer and flow redistribution within this range of reactor operation.
- 1677 (HEDL-TI--45-LMB-009) EXPERIMENTAL RESULTS OF SODIUM-WATER REACTION TEST NO. 3 IN LLTR. Miller, A.E.; Neely, H.H. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 1 Aug 1977. Contract EY-76-C-03-0700. 569p. TIC \$16.25.
- Computer-generated plots of the transient data obtained during the third sodium-water reaction test (SWR-3) and observations made after the test are presented. Similar to the first two tests of Series I, a double-ended guillotine rupture was produced in a water tube of the Atomics International Modular Steam Generator (AI-MSG). Prior to tube rupture, the temperature distribution in the vertical AI-MSG was linear from 600°F at the bottom to 800°F at the top. The rupture was located in the horizontal section 1.75 in. from the upper tubesheet. Peak pressures generated in this test were somewhat lower than the 400 psi and 500 psi measured in the prior tests; while peak temperatures, about 1600°F, were higher than were measured previously. The interest examinations revealed no structural damage, material wastage, stress corrosion cracking, or dimensional changes. Additional reaction products have been accumulated in the bottom of the AI-MSG up to Spacer 3, so that the flow path in the AI-MSG to both the lower relief line and the drain line is restricted. The relief lines are relatively clear.
- 1678 (HEDL-TI--45-LME-006) EXPERIMENTAL RESULTS OF SODIUM-WATER REACTION TEST NO. 1 IN LLTR. Leppard, J.A.; Neely, H.H. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 15 Dec 1976. Contract EY-76-C-03-0700. 540p. AT.
- Sodium-Water Reaction Test No. 1 (SWR-1), the first test conducted in the Large-Leak Test Rig (LLTR), is described. The test consisted of intentionally producing a double-ended guillotine rupture of a tube in the Atomics International Modular Steam Generator, which had been modified for large-leak testing subsequent to performance tests in the Sodium Components Test installation. Results of post-test inspections of the test steam generator are discussed. Graphical displays of test data, which are to be used to validate or modify analytical techniques for predicting large-leak effects, are presented.
- 1679 (HEDL-TI--45-LME-007) EXPERIMENTAL RESULTS OF SODIUM-WATER REACTION TEST NO. 2 IN LLTR. Adler, K.L.; Neely, H.H. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 10 May 1977. Contract EY-76-C-03-0700. 538p. TIC \$15.50.
- The second sodium-water reaction test, SWR-2, in Series I of the Large Leak Test Rig (LLTR) program is described. Like the first test, SWR-1, this test consisted of intentionally producing a double-ended guillotine rupture of a single tube in the Atomics International-Modular Steam Generator (AI-MSG). The location of the SWR-2 rupture was higher and radially nearer the center-line of the steam generator than the SWR-1 rupture. Pressures and temperatures produced in SWR-2 were similar to those in SWR-1. This report includes results of post-test inspections of the test article and graphical presentations of the test data.
- 1680 (HEDL-TI--75001-9) REVISIONS TO FFTF FSAR. SUPPLEMENT 9, PARTS 1, 2, 3, 4, AND 5. (Hanford Engineering Development Lab., Richland, WA (USA)). 10 Apr 1978. Contract EY-76-C-14-2170. 63p. AT.
- Supplementary information is presented for incorporation into the FFTF Final Safety Analysis Report. (DG)
- 1681 (HEDL-TI--75001-10) FSAR Q₁ RESPONSES. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1977. Contract EY-76-C-14-2170. 67p. AT.
- Supplementary information related to the FFTF final Safety Analysis Report is presented. (DG)
- 1682 (HEDL-TI--75001-11) FSAR Q₁ RESPONSES. SUPPLEMENT 11. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1977. Contract EY-76-C-14-2170. 82p. AT.
- Additional information is presented on analytical methods and data used in the FFTF Final Safety Analysis Report. (DG)
- 1683 (HEDL-TI--75001-12-1) FSAR Q₁ RESPONSES SUPPLEMENT 12. PART 1. (Hanford Engineering Development Lab., Richland, WA (USA)). 1977. Contract EY-76-C-14-2170. 90p. AT.
- Information is presented concerning the performance characteristics of FFTF fuel pins.
- 1684 (HEDL-TI--75001-12.2) FSAR SUPPLEMENT 12, PART 2, RESPONSES TO NRC Q₁-SET 7; AMENDMENT 13. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 147p. AT.
- Requested additional information on the FFTF reactor is presented concerning fuel pin performance limits and cladding failure.
- 1685 (HEDL-TI--75001-13.1) FSAR SUPPLEMENT 13. PART I. RESPONSES TO NRC Q₁ SET 8. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 65p. AT.
- Requested additional information on the FFTF reactor is presented concerning reactor characteristics following control rod withdrawal accidents, loss of power, and loss of flow accidents.
- 1686 (HEDL-TI--75001-14) FSAR SUPPLEMENT 14: RESPONSE TO NRC Q₁ SET 9. Schade, A.R.

(Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 19p. AT.

Requested additional information on the FFTF reactor is presented concerning personnel qualifications and staffing requirements.

- 1687 (HEDL-TI--75001-15) RESPONSES TO NRC Q₂ SET 3 QUALITY ASSURANCE. FSAR SUPPLEMENT 15. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1977. Contract EY-76-C-14-2170. 40p. AT.

Responses to NRC questions regarding the quality assurance program for the FFTF reactor are presented.

- 1688 (HEDL-TI--75001-16.1) DISTRIBUTION OF REVISIONS TO FFTF FSAR. (Hanford Engineering Development Lab., Richland, WA (USA)). 24 Feb 1978. Contract EY-76-C-14-2170. 23p. AT.

Supplementary information is presented for incorporation into the FFTF Final Safety Analysis Report. The supplement deals with estimation of the maximum atmospheric particulate concentrations and particulate size distribution during periods of extreme atmospheric dust loadings in the site area. (DG)

- 1689 (HEDL-TI--75001-16.2) FSAR SUPPLEMENT 16. PART 2. RESPONSE TO NRC Q₂ SET 2. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 28p. AT.

Requested additional information on the FFTF reactor is presented concerning personnel qualifications and testing of diesel generators, instrumentation, and sodium leak detection system.

- 1690 (HEDL-TI--75001-17) FSAR SUPPLEMENT 17: RESPONSES TO NRC Q₁ SET 10. (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1977. Contract EY-76-C-14-2170. 84p. AT.

Requested additional information on the FFTF reactor is presented concerning the effects of thermal stresses on the steel vessel liner.

- 1691 (HEDL-TI--75001-18) FSAR SUPPLEMENT 18: RESPONSES TO NRC Q₁ SET 6. Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1977. Contract EY-76-C-14-2170. 137p. AT.

Requested additional information on the FFTF reactor is presented concerning electrical equipment, instrumentation, and reactor protection systems.

- 1692 (HEDL-TI--75001-19) FSAR SUPPLEMENT 19: RESPONSE TO NRC Q₂ SET 3. Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 72p. AT.

Requested additional information on the FFTF reactor is presented concerning stress analysis of piping systems and core restraint system.

- 1693 (HEDL-TI--75001-20) FSAR SUPPLEMENT 20, RESPONSE TO NRC Q₂ SET 4. Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 70p. AT.

Additional information concerning the FFTF Final Safety Analysis Report is presented in response to NRC questions. The information covers accidental radioactivity release and

exposure data.

- 1694 (HEDL-TI--75001-21) FSAR SUPPLEMENT 21: RESPONSE TO NRC Q₂ SET 5. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 47p. AT.

Requested additional information on the FFTF reactor is presented concerning seismic effects on structures and stress analysis of buildings and structural materials.

- 1695 (HEDL-TI--75001-22) FSAR SUPPLEMENT 22: RESPONSES TO NRC Q₂ SET 6. Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 27p. AT.

Requested additional information on the FFTF reactor is presented concerning containment leak testing and specifications for containment system valves.

- 1696 (HEDL-TI--75001-23) FSAR SUPPLEMENT 23: RESPONSES TO NRC Q₂ SET 7. Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 66p. AT.

Requested additional information on the FFTF reactor is presented concerning reactor materials monitoring; inspection of components and welded joints; and sodium leak detectors.

- 1697 (HEDL-TI--75001-24) RESPONSE TO NRC Q₂ SET 8. FSAR SUPPLEMENT 24. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 12p. AT.

Responses to NRC questions concerning the SPRAY Code described in the FFTF FSAR are presented.

- 1698 (HEDL-TI--75001-25) DISTRIBUTION OF REVISIONS TO FFTF FSAR. Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). 16 Dec 1977. Contract EY-76-C-14-2170. 30p. AT.

Supplementary information related to FFTF primary and secondary isolation valve control power sources is presented for incorporation into the FFTF Final Safety Analysis Report. (DG)

- 1699 (HEDL-TI--75001-26) FSAR SUPPLEMENT 26: RESPONSE TO NRC Q₂ SET 11. (Hanford Engineering Development Lab., Richland, WA (USA)). 9 Mar 1978. Contract EY-76-C-14-2170. 54p. AT.

Supplementary information is provided for incorporation in the FFTF Final Safety Analysis Report. (DG)

- 1700 (HEDL-TI--75001-27) FSAR SUPPLEMENT 27: RESPONSE TO NRC Q₂ SET 12. (Hanford Engineering Development Lab., Richland, WA (USA)). 9 Mar 1978. Contract EY-76-C-14-2170. 25p. AT.

Supplementary information is presented for incorporation in the FFTF Final Safety Analysis Report.

- 1701 (HEDL-TI--75001-28) FSAR SUPPLEMENT 28: RESPONSE TO NRC Q₂ SET 13. (Hanford Engineering Development Lab., Richland, WA (USA)). 9 Mar 1978. Contract EY-76-C-14-2170. 19p. AT.

Supplementary information is provided for

- incorporation in the FFTF Final Safety Analysis Report. (DG)
- 1702 (HEDL-TI--75001-29) FSAR SUPPLEMENT 29: RESPONSE TO NRC Q₂ SET 14. (Hanford Engineering Development Lab., Richland, WA (USA)). [nd]. Contract EY-76-C-14-2170. 31p. AT.
Responses to questions concerning the FFTF safety analysis are presented.
- 1703 (HEDL-TI--75001-30) FSAR SUPPLEMENT 30. (Hanford Engineering Development Lab., Richland, WA (USA)). 9 Oct 1978. Contract EY-76-C-14-2170. 63p. AT.
Information on the FFTF reactor is presented concerning instrumentation, reactor protection systems, and performance of engineered safety systems.
- 1704 (HEDL-TI--75002-5) FSAR AMENDMENT 5, REVISIONS TO SUPPLEMENT 6. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1977. Contract EY-76-C-14-2170. 16p. AT.
Information on the FFTF Reactor is presented concerning the performance of structures under tornado and seismic stresses, securing of reactor internals, and reactor kinetics.
- 1705 (HEDL-TI--75002-8) FSAR AMENDMENT 8. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1977. Contract EY-76-C-14-2170. 91p. AT.
Revisions to the FFTF Final Safety Analysis Report are presented. (DG)
- 1706 (HEDL-TI--75002-9) FFTF EMERGENCY PLAN, FSAR AMENDMENT 9. (Hanford Engineering Development Lab., Richland, WA (USA)). 10 Apr 1978. Contract EY-76-C-14-2170. 147p. AT.
The emergency plan for the FFTF reactor is presented.
- 1707 (HEDL-TI--75002-10) FSAR AMENDMENT 10. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1977. Contract EY-76-C-14-2170. 87p. AT.
Supplementary information is provided on the FFTF Final Safety Analysis Report. (DG)
- 1708 (HEDL-TI--75002-11) FSAR AMENDMENT 11: CHANGES TO REFLECT SUPPLEMENT 13, PART 1. Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 12p. AT.
Revised information is presented concerning FFTF reactor operating procedures.
- 1709 (HEDL-TI--75002-12) DISTRIBUTION OF REVISIONS TO FFTF FSAR. Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). 9 Jan 1978. Contract EY-76-C-14-2170. 50p. AT.
Supplementary information for incorporation into the FFTF final Safety Analysis Report is presented. (DG)
- 1710 (HEDL-TI--75002-14) FSAR AMENDMENT 14, FSAR REVISIONS: RESPONSE TO NRC Q₂ SET 1, QUALITY ASSURANCE. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1977. Contract EY-76-C-14-2170. 32p. AT.
An amended Quality Assurance Chapter is presented for the FFTF Final Safety Analysis
- Report. (DG)
- 1711 (HEDL-TI--75002-15) FSAR AMENDMENT 15: MISCELLANEOUS CHANGES. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 75p. AT.
Requested additional information on the FFTF reactor is presented concerning the health physics program and radiation protection equipment and procedures.
- 1712 (HEDL-TI--75002-16) FSAR AMENDMENT 16. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 21p. AT.
Revisions to the FFTF Final Safety Analysis Report are presented. The revisions concern sodium release in secondary systems, engineered safety feature instrumentation, and containment isolation valves.
- 1713 (HEDL-TI--75002-17) FSAR AMENDMENT 17. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 15p. AT.
Revised data concerning load combinations and allowable stresses on containment structures are presented for addition to the FFTF Final Safety Analysis Report.
- 1714 (HEDL-TI--75002-18) DISTRIBUTION OF REVISIONS TO FFTF FSAR. Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). 16 Dec 1977. Contract EY-76-C-14-2170. 16p. AT.
Requested additional information on the FFTF reactor is presented concerning containment system valves.
- 1715 (HEDL-TI--75002-19) DISTRIBUTION OF REVISIONS TO FFTF FSAR. Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). 9 Jan 1978. Contract EY-76-C-15-2170. 15p. AT.
Supplementary information for incorporation into the FFTF Final Safety Analysis Report is presented. The amendment deals specifically with seismic qualification of safety-related valves. (DG)
- 1716 (HEDL-TI--75002-20) DISTRIBUTION OF REVISIONS TO FFTF FSAR. Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). 9 Jan 1978. Contract EY-76-C-14-2170. 32p. AT.
Supplementary information for incorporation into the FFTF Final Safety Evaluation Report is presented. The amendment deals specifically with fire protection changes. (DG)
- 1717 (HEDL-TI--75002-21) FFTF FSAR, AMENDMENT 21. (Hanford Engineering Development Lab., Richland, WA (USA)). 8 Mar 1978. Contract EY-76-C-14-2170. 33p. AT.
Miscellaneous revisions to the FFTF Final Safety Analysis Report are presented.
- 1718 (HEDL-TI--75002-22) FSAR AMENDMENT 22. ADDITION OF RESPONSE Q130.57 TO SUPPLEMENT 21. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1978. Contract EY-76-C-14-2170. 14p. AT.
Additional data on the seismic analysis of FFTF structure is presented in response to an NRC request.

1719 (HEDL-TI--75002-23) FSAR AMENDMENT 23: (CHANGE TO CHAPTER 17, AS MADE BY FSAR SUPPLEMENT 26). (Hanford Engineering Development Lab., Richland, WA (USA)). 9 Mar 1978. Contract EY-76-C-14-2170. 13p. AT. Supplementary information is provided for incorporation in the FFTF Final Safety Analysis Report. (DG)

1720 (HEDL-TI--75002-24) FSAR AMENDMENT 24: ADDITION TO SUPPLEMENT 13, Q211.33, NRC Q₁, SET 8. (Hanford Engineering Development Lab., Richland, WA (USA)). 9 Mar 1978. Contract EY-76-C-14-2170. 25p. AT. Supplementary information is provided for incorporation in the FFTF Final Safety Analysis Report. (DG)

1721 (HEDL-TI--75002-25) FSAR AMENDMENT 25: MISCELLANEOUS REVISIONS. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1978. Contract EY-76-C-14-2170. 129p. AT. Supplementary information is presented for incorporation in the FFTF Final Safety Analysis Report. (DG)

1722 (HEDL-TI--75002-26) DISTRIBUTION OF REVISIONS TO FFTF FSAR. (Hanford Engineering Development Lab., Richland, WA (USA)). 1 Sep 1978. Contract EY-76-C-14-2170. 53p. AT. Supplementary information is presented for inclusion in the FFTF Final Safety Analysis Report.

1723 (HEDL-TI--75002-28) FSAR AMENDMENT 28. (Hanford Engineering Development Lab., Richland, WA (USA)). 9 Oct 1978. Contract EY-76-C-14-2170. 24p. AT. Revised information on the FFTF reactor is presented concerning instrumentation, reactor protection systems, and engineered safety systems.

1724 (HEDL-TI--75002-29) FSAR AMENDMENT 29: MISCELLANEOUS CHANGES. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1979. Contract EY-76-C-14-2170. 265p. AT. Portions of document are illegible. Revised and additional information is presented concerning the reactor control system; containment isolation system; plant protection systems; electrical systems; plant fire protection; in-vessel handling machine; and safety limit settings for engineered safeguards.

1725 (HEDL-TI--75002-30) REVISION OF FFTF FSAR TECHNICAL SPECIFICATIONS. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1979. Contract EY-76-C-06-1830. 266p. AT. Revised technical specifications are presented for three-loop operation of the FFTF at 680°F inlet conditions with cores 1 and 2 and without closed-loop experiments in the reactor.

1726 (HEDL-TI--75002-31) FSAR AMENDMENT 31: CHANGES TO SECTION 3.10. Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1979. Contract EY-76-C-14-2170. 75p. AT. Revisions to Section 3.10 (Environmental Design of Mechanical and Electrical Equipment) of the FFTF Final Safety Analysis Report are presented.

1727 (HEDL-TI--75002-32) FSAR AMENDMENT 32: MISCELLANEOUS CHANGES. Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1979. Contract EY-76-C-14-2170. 87p. AT. FFTF REACTOR; REACTOR SAFETY; DOCUMENTATION

1728 (HEDL-TI--75002-33) DISTRIBUTION OF REVISIONS TO FFTF FSAR. (Hanford Engineering Development Lab., Richland, WA (USA)). 12 Mar 1979. Contract EY-76-C-14-2170. 56p. AT. Requested additional information is presented concerning the reactor protection systems and auxiliary equipment.

1729 (HEDL-TI--75002-34) FSAR AMENDMENT 34: ENVIRONMENTAL TECHNICAL SPECIFICATIONS FOR THE FAST FLUX TEST FACILITY (CHAPTER 18). Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1979. Contract EY-76-C-14-2170. 67p. AT. Requested additional information is presented concerning environmental surveillance procedures.

1730 (HEDL-TI--75002-35) FFTF FSAR AMENDMENT 35. REVISIONS TO EMERGENCY PLAN. (Hanford Engineering Development Lab., Richland, WA (USA)). 18 Apr 1979. Contract EY-76-C-14-2170. 53p. AT. Supplementary information is presented for inclusion in the FFTF Final Safety Analysis Report.

1731 (HEDL-TI--76002) SODIUM. (Westinghouse Hanford Co., Richland, WA (USA)). [nd]. Contract EY-76-C-14-2170. 4p. Dep. NTIS, PC A02/MF A01. The physical properties of sodium are described and the use of sodium as the coolant in liquid metal fast breeder reactors is discussed. Sodium technology facilities at HEDL for research and testing support of the Fast Flux Test Facility are described.

1732 (HEDL-TI-LME--012) EXPERIMENTAL RESULTS OF SODIUM--WATER REACTION TESTS NOS. 4 AND 5 IN LLTR. Miller, A.E.; Neely, H.H. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 10 Nov 1977. Contract EY-76-C-03-0700. 815p. AT. Computer-generated plots of transient data obtained during the fourth and fifth (SWR-4 and -5) sodium-water reaction tests and observations made following the tests are presented. As in previous tests of Series I, double-ended guillotine ruptures were produced in a water tube of the Atomics International Modular Steam Generator (AI-MSG). Tube rupture was located in the horizontal section 1.75 in. from the upper tube-sheet for each test. Superheated steam at approximately 700°F and 1900 psig was supplied for injection at tube rupture in SWR-4; whereas, high-pressure nitrogen was injected in SWR-5. Comparison of peak pressure data from the two tests shows similarity that may indicate system water pressure determines peak pressure.

1733 (HEDL-TME--72-86) EFFECT OF LIQUID SODIUM ON FFTF IHX TUBE-TO-TUBESHEET WELD RESIDUE DEPOSITS. Yunker, W.H. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1972. Contract EY-76-C-14-2170. 24p. Dep. NTIS, PC A02/MF A01. Black deposits, about 20 mil diameter and 5

mil thick, formed during tube-to-tubesheet weld qualification tests for the FFTF Intermediate Heat Exchanger. The behavior of these spots in liquid sodium has been studied in the temperature range 400°F to 800°F in flowing and in stagnant sodium. The results show that the spot-like deposits disintegrate in a few hours into many smaller pieces and do not come loose from the metal as single pieces. The dissolution/disintegration can be assumed to continue after the fragments are released into the sodium. The total mass of material released is estimated to increase the impurity level in the FFTF primary sodium by 0.004 wppm.

- 1734 (HEDL-TME--72-95) LENGTH CHANGES IN FTR PROTOTYPIC CLADDING IRRADIATED IN EBR-II TO 1×10^{22} N/CM², $E > 0.1$ MEV (PIN B-65)(RM-24). Straalsund, J.L.; Guthrie, G.L.; Larson, T.J. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1972. Contract EY-76-C-14-2170. 50p. Dep. NTIS, PC A04/MF A01.

Swelling in prototypic FTR cladding and duct specimens is being determined from a series of three irradiations designed to cover a range of temperatures approximately 800 to 1500°F and peak fluences of approximately 1×10^{22} , 6×10^{22} , and 1×10^{23} N/cm². The length change measurements for the low fluence irradiation have been completed and are summarized in this report. No swelling, only shrinkage, was found in the 20 percent cold worked Type 316 stainless steel cladding specimens. This shrinkage is of the same magnitude and has the same temperature dependency as unirradiated, control samples. Annealed specimens did swell slightly. Details of the experiment and analysis are given.

- 1735 (HEDL-TME--75-85) TEMP: A COMPUTER CODE TO CALCULATE FUEL PIN TEMPERATURES DURING A TRANSIENT. Bard, F.E.; Christensen, B.Y.; Gneiting, B.C. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1980. Contract EY-76-C-14-2170. 293p. NTIS, PC A13/MF A01.

The computer code TEMP calculates fuel pin temperatures during a transient. It was developed to accommodate temperature calculations in any system of axis-symmetric concentric cylinders. When used to calculate fuel pin temperatures, the code will handle a fuel pin as simple as a solid cylinder or as complex as a central void surrounded by fuel that is broken into three regions by two circumferential cracks. Any fuel situation between these two extremes can be analyzed along with additional cladding, heat sink, coolant or capsule regions surrounding the fuel. The one-region version of the code accurately calculates the solution to two problems having closed-form solutions. The code uses an implicit method, an explicit method and a Crank-Nicolson (implicit-explicit) method.

- 1736 (HEDL-TME--75-96) FTR HCM SCALE-MODEL FLOW-INDUCED VIBRATION TESTS, PHASES I AND II. FINAL REPORT. Ryan, J.A.; Mahoney, J.J. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1979. Contract EY-76-C-14-2170. 116p. AT.

The flow-induced vibrational characteristics of scaled Fast Flux Test Facility (FFTF) Fast Test Reactor (FTR) core internal and peripheral components were assessed under scaled and simulated FTR flow conditions in the Hydraulic Core Mockup (HCM). The HCM was an 0.285 geometric scale model of the FTR internals and, as such, was designed to model the vibrational and hydraulic characteristics of the FTR.

Vibration tests were conducted with water simulating the coolant over the range of 36 to 110% scaled FTR design flow rates. Selected model internals were instrumented with accelerometers and their vibrational characteristics were measured and established for the range of flow rates. The measured dynamic response of the scaled FTR components exhibited neither abnormal nor anomalous flow rate dependent response. Component vibrational characteristics were adjudged acceptable in terms of FTR response.

- 1737 (HEDL-TME--76-1) SCALE-MODEL CHARACTERIZATION OF FLOW-INDUCED VIBRATIONAL RESPONSE OF FTR INTERNALS: SUMMARY REPORT. Ryan, J.A.; Julyk, L.J. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1977. Contract EY-76-C-14-2170. 70p. AT.

As an integral part of the Fast Test Reactor (FTR) Vibration Program for Reactor Internals, the flow-induced vibrational characteristics of scaled FTR core internal and peripheral components were assessed under scaled and simulated prototype flow conditions in the Hydraulic Core Mockup (HCM). The HCM, a 0.285 geometric scale (water) model, was designed to simulate the vibrational and hydraulic characteristics of the FTR. Model component vibrational characteristics were measured and determined over a range of 36 to 110% of the scaled prototype design flow. Selected model and prototype components were shaker tested to establish modal characteristics. The dynamic response of the HCM components exhibited no anomalous flow rate dependent or modal characteristics, and prototype response predictions were adjudged acceptable.

- 1738 (HEDL-TME--76-3) COVER GAS ENTRAINMENT TESTS IN FFTF HYDRAULIC CORE MOCKUP. Dickinson, D.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1978. Contract EY-76-C-14-2170. 54p. AT.

Tests were performed in a 0.285-scale model to determine the rate of cover gas entrainment that might be expected in the outlet plenum of the Fast Flux Test Facility due to upward flow from the reactor core. Water was used to model sodium. The major source of entrainment was the gap surrounding a large vertical column where it penetrated the horizontal suppressor plate below the liquid surface. Modifications of this gap were tested to support redesign to reduce entrainment. The void fraction in the outlet pipes was measured by recording the attenuation of an ultrasonic pulse transmitted across the pipe, the system was calibrated in place by measured gas flow. The outlet void fraction for the final modification was 6×10^{-4} at Weber number modeling, which gives an upper bound on FFTF performance. The outlet void fraction was zero at Froude number modeling.

- 1739 (HEDL-TME--76-4) LIQUID VELOCITY MEASUREMENTS IN OUTLET PLENUM OF FFTF HYDRAULIC CORE MOCKUP. Dickinson, D.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1977. Contract EY-76-C-14-2170. 87p. AT.

Liquid velocities were measured in the outlet plenum of the Hydraulic Core Mockup, which is a 0.285-scale hydraulic water model of the Fast Test Reactor, using three-dimensional pitot probes. Velocities (at FTR conditions) were 6 to 13 ft/sec upward over the core, 1 to 2 ft/sec radially outward below the vortex suppression plate, and 1 to 3 ft/sec downward along the vessel wall.

- 1740 (HEDL-TME--76-5) TESTS OF FLUID MIXING IN INLET AND OUTLET PLENUMS OF FFTF HYDRAULIC CORE MOCKUP. Dickinson, D.R. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1979. Contract EY-76-C-14-2170. 142p. AT.

Mixing sodium streams of different temperatures in the inlet and outlet plenums of the Fast Flux Test Facility reactor was evaluated by tests in water in a 0.285-scale hydraulic model of the reactor vessel. Flow streams of water were tagged by injecting concentrated electrolyte to model hot or cool sodium streams. The resulting concentration changes and concentration fluctuations were measured by 162 small probes which sense local electrical conductivity. Measurements were made for mixing in the inlet plenum following stepwise concentration increase in one of the inlet pipes, and also for mixing in the outlet plenum following stepwise concentration increase in core positions within a single orifice zone. The inlet plenum was found to be well mixed; the outlet plenum less so. Data are presented which can be used to predict reactor steady-state temperatures, temperature transients, and temperature fluctuations.

- 1741 (HEDL-TME--76-10) ACTIVATED CORROSION PRODUCT RADIATION LEVELS IN THE FFTF HEAT TRANSPORT SYSTEM CELLS AND CLOSED LOOP SYSTEM MODULES. Kuhn, W.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1977. Contract EY-76-C-14-2170. 115p. AT.

Radiation levels due to activated corrosion products in an FFTF Heat Transport System (HTS) cell and in a Closed Loop System (CLS) module are estimated for various maintenance locations. ^{54}Mn is found to be the dominant radionuclide. While control of ^{54}Mn release or transport would substantially reduce radiation levels, ^{54}Mn , ^{60}Co , and ^{59}Co must all be controlled to permit unshielded maintenance for the FFTF rated operating condition. Radiation levels are found to be relatively insensitive to sodium oxygen concentration but very sensitive to operating temperature. The average radiation level in an HTS cell is predicted to rapidly exceed 0.2 R/hr and eventually reach 4 R/hr at the FFTF rated core outlet temperature of 1100°F. At the 980°F, 0.7 R/hr is interpolated. The predicted radiation levels would preclude unshielded maintenance operations in an HTS cell. Radiation levels near a CLS module are predicted 0.2 to 2 R/hr, depending on the assumed operating temperature.

- 1742 (HEDL-TME--76-28) SODIUM ENGINEERING AND TECHNOLOGY TECHNICAL PROGRESS REPORT, OCTOBER, NOVEMBER, DECEMBER 1976. Atwood, J.M. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1977. Contract EY-76-C-14-2170. 112p. AT.

Progress in LMFBF-related sodium technology studies is reported in the areas of radioactivity control technology, sodium systems technology, and hydraulics and mechanics studies.

- 1743 (HEDL-TME--76-45) FFTF DEVELOPMENT TESTING TECHNICAL PROGRESS REPORT, JANUARY, FEBRUARY, MARCH 1976. Berg, J.D.; Bliss, R.J.; Duncan, H.C.; Worth, G.A.; Duckett, A.J. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1977. Contract EY-76-C-14-2170. 24p. AT.

Information is presented concerning refueling and plant equipment testing, mechanical equipment testing, reactor components testing, and control and data systems.

- 1744 (HEDL-TME--76-46) FFTF DEVELOPMENT TESTING TECHNICAL PROGRESS REPORT, APRIL--JUNE 1977. Berg, J.D.; Bliss, R.J.; Duckett, A.J.; Duncan, H.C.; Worth, G.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1977. Contract EY-76-C-14-2170. 50p. AT.

Progress in FFTF component development is presented for the following: refueling and plant equipment testing, mechanical equipment testing, reactor components testing, and control and data systems.

- 1745 (HEDL-TME--76-47) FFTF DEVELOPMENT TESTING. TECHNICAL PROGRESS REPORT, JULY--SEPTEMBER 1976. Bliss, R.J. (Hanford Engineering Development Lab., Richland, WA (USA)). 1976. Contract EY-76-C-14-2170. 32p. AT.

Progress is summarized in the following areas: (1) refueling and plant equipment testing, (2) mechanical equipment testing, and (3) control and data systems. (DG)

- 1746 (HEDL-TME--76-48) FFTF DEVELOPMENT TESTING TECHNICAL PROGRESS REPORT, OCTOBER--DECEMBER 1976. Berg, J.D.; Bliss, R.J.; Duckett, A.J.; Duncan, H.C.; Worth, G.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1977. Contract EY-76-C-14-2170. 30p. AT.

Progress in FFTF development testing is presented for the following: refueling and plant equipment testing, mechanical equipment testing, control and data systems, and reactor components testing.

- 1747 (HEDL-TME--76-65) FFTF TORNADO MISSILE BARRIER REPORT. Shang, K.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1977. 36p. AT.

An evaluation is presented of the series of tests that was performed by the Sandia Corporation to determine the sufficiency of several systems that are candidates for barriers against tornado generated missiles for FFTF structures. Several objects representative of tornado missiles were impact tested against single and multiple layers of chain link fence and a concrete block wall. The barrier systems are being considered as alternatives to steel plate tornado hardening of the FFTF Dump Heat Exchangers and the containment airlocks.

- 1748 (HEDL-TME--77-13) DECAY HEAT FOR THE FAST TEST REACTOR (FTR). Schmittroth, F. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1978. Contract EY-76-C-14-2170. 50p. Dep. NTIS, PC A03/MF A01.

The decay heat for the Fast Test Reactor (FTR) is reevaluated for cooling times up to 30 years. Decay heat from both fission-products and actinides is included. Fission-product decay data used in this evaluation are taken from ENDF/B version IV. Consideration is given to the effects of fuel cycles and loadings and to the importance of fissions in the minor fuels such as ^{235}U , ^{240}Pu , and ^{241}Pu , as well as in the major fuels, ^{238}U and ^{239}Pu . Decay-heat uncertainties are evaluated and are typically less than 5 percent for cooling times beyond 50 seconds. The effective energy release per fission for FTR is also reevaluated.

- 1749 (HEDL-TME--77-20) THERMAL PERFORMANCE TESTS OF THE FFTF CLOSED LOOP COLD TRAP.

Blair, C.K.; Bloom, G.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1977. Contract EY-76-C-14-2170. 100p. AT.

The three gpm FFTF Closed Loop Cold Trap was tested to demonstrate thermal performance of the economizer and gas cooled crystallizer. Sodium and gas stream temperature profiles were measured and confirmed with heat balance calculations over the expected operating range. The trace heat system design adequacy was also tested.

Results of the effluent releases from the HEDL Fuel Fabrication Demonstration building are given; also presented are effluent measurements from a joint program with Babcock and Wilcox at the Fuel Fabrication Facility at Leechburg, PA. Attenuation measurements of flexible shielding materials designated 'Lamipore' are given and the neutron and gamma radiation levels from FFTF subassemblies and strip layers are recorded. A specific use of a Stabilized Assay Meter (SAM-2) for detecting plutonium in a waste dissolver is recorded.

1750 (HEDL-TME--77-23) INTEGRAL HEAT RATE-TO-INCIPENT MELTING IN UO_2 - PuO_2 FAST REACTOR FUEL. Baker, R.B. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1978. Contract EY-76-C-14-2170. 132p. Dep. NTIS, PC A07/MF A01.

Analyses of integral power-to-melt tests, HELD P-19 and P-20, are presented. These tests contained mixed-oxide, UO_2 - PuO_2 , fuel prototypic of fuel to be used in the driver fuel assemblies of the Fast Flux Test Facility, FFTF, which could be used in initial Fast Breeder Reactors, FBR's. The experiments were irradiated in the Experimental Breeder Reactor No. 2, EBR-II, for the Hanford Engineering Development Laboratory, HELD. The primary test variables are fabricated fuel-to-cladding gap size and accumulated fuel burnup to 1.1 at.%. The observed improved thermal performance at higher burnup is analyzed based on results and conclusions drawn from sibling pins irradiated during the preconditioning phases of the P-20 test. These pins were not included in the overpower phase of the P-20 test. The primary cause of the improved thermal performance was found to be maturing fuel restructuring and rapid fuel-to-cladding gap closure. The favorable impact of these observations on the design and operation of FFTF and future FBR's is also reviewed.

1751 (HEDL-TME--77-25) IMPROVED SUBASSEMBLY GROUPING FOR MULTI-CHANNEL TOP HCDA SAFETY ANALYSIS. Wilburn, N.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1977. Contract EY-76-C-14-2170. 50p. AT.

The behavior of two FTR cores, the BOC-4 and EOC-4, is investigated for TOP HCDA's using the 100-channel MELT-IIIA code. Each of the 76-driver subassemblies is assigned a channel. Using the results from the 76-channel runs, a more effective method to group subassemblies into a lesser number of channels is developed. The results of a 7-channel grouping using this method are compared to results from a former method and the 76-channel runs.

1752 (HEDL-TME--77-29) ROD SCANNER ASSAY OF FFTF FUEL. Goris, P. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1978. Contract EY-76-C-14-2170. 29p. Dep. NTIS, PC A03/MF A01.

Rod scanner results for over 40,000 FFTF Cores 1 and 2 fuel pins are summarized. An improved rod scanner calibration technique for FFTF Cores 3 and 4 fuel pins is described. Data are compared with chemical assay measurements. 5 fig., 5 tables.

1753 (HEDL-TME--77-30) RADIATION AND EFFLUENT CONTROL TECHNOLOGY SEMI-ANNUAL PROGRESS REPORT, JANUARY--JUNE 1976. Smith, R.C.; Strode, J.N.; Bradley, J.G.; Richardson, G.L.; Brackenbush, L.W.; Faust, L.G.; Mishima, J. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1977. Contract EY-76-C-14-2171. 46p. AT.

1754 (HEDL-TME--77-33) LMFBR REFERENCE CONTROL MATERIALS. SEMI-ANNUAL REPORT, JULY--DECEMBER 1976. Roake, W.E.; Claudson, T.T. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1977. Contract EY-76-C-14-2170. 67p. AT.

Information is presented concerning improved absorber concepts, design and performance analysis, materials testing and characterization, neutron absorber performance monitoring, and fabrication development.

1755 (HEDL-TME--77-34(Vol.2), pp HEDL.60-HEDL.65) MODIFICATION OF THE NBS DOUBLE FISSION CHAMBER FOR IN-CORE OPERATION IN THE FAST TEST REACTOR. Fuller, J.L. May 1978. Interlaboratory Reaction Rate program. 11th progress report, November 1975--October 1976.

The results of studies undertaken to determine the design changes required to the National Bureau of Standards double fission chamber for use in FTR are provided. The detector design environment parameters are long-term operation at 400°F, γ -field to 1×10^7 R/h, 1 g vibration amplitude from 10 to 500 Hz, and neutron flux to 7×10^{12} cm⁻² s⁻¹. The status of efforts to implement the necessary modifications is also briefly discussed. 2 figures.

1756 (HEDL-TME--77-34(Vol.2), pp HEDL.27-HEDL.44) EBR-II DERIVED FISSION YIELDS FROM VANADIUM ENCAPSULATED FISSION MONITORS. Kellogg, L.S.; Davis, A.I.; Lippincott, E.P.; Ruggles, J.M. May 1978. Interlaboratory Reaction Rate program. 11th progress report, November 1975--October 1976.

Relative fission yield measurements were made for selected fission products obtained from irradiation of encapsulated ²³⁵U, ²³⁸U, ²³⁹Pu, and ²³⁷Np dosimeters in Row 4 'Test Region' of the EBR-II reactor. Replicate dosimeter sets were positioned in three separate locations to determine consistency of analysis, validity of fission yields recommended by the ILRR Program and elsewhere for use in FFTF and EBR-II, and to obtain data on possible spectral/energy dependence of the fission product yields. Selection of the fission products reported was based on those products routinely used for dosimetry foil and burnup analysis. Consistency of within-group radiometric analysis, with few exceptions, is quite good and well within the accuracy estimates of the individual analysis. Tabulations of individual yield to ¹³⁷Cs ratios are made. Comparison of fission product/¹³⁷Cs ratios with ratio values derived from ILRR consensus fission yield values agree within 2.5% with the exception of ⁹⁵Zr/²³⁷Np=6.6%, ¹⁰³Ru/²³⁸U=4.7%, and ¹⁰³Ru/²³⁷Np=3.4%. An apparent consistent trend in the ¹⁰³Ru ratios from ²³⁵U would suggest an energy dependence, but the comparative data are not statistically significant for confirmation.

- 1757 (HEDL-TME--77-36) SODIUM ENGINEERING AND TECHNOLOGY. QUARTERLY PROGRESS REPORT, JANUARY, FEBRUARY, MARCH 1977. Atwood, J.M. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1977. Contract EY-76-C-14-2170. 107p. AT.

Progress is reported in the areas of: radioactivity control technology, sodium systems technology, sodium systems development, and hydraulics and mechanics. (GHT)

- 1758 (HEDL-TME--77-37) SODIUM ENGINEERING AND TECHNOLOGY. TECHNICAL PROGRESS REPORT, APRIL, MAY, JUNE 1977. Atwood, J.M. (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1977. Contract EY-76-C-14-2170. 142p. AT.

Progress in LMFBR-related sodium technology studies is reported in the areas of radioactivity control technology, sodium systems technology, safety systems development, and the hydraulics and mechanics program.

- 1759 (HEDL-TME--77-38) SODIUM ENGINEERING AND TECHNOLOGY. TECHNICAL PROGRESS REPORT, JULY, AUGUST, SEPTEMBER 1977. Atwood, J.M. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 102p. AT.

Progress in LMFBR-related sodium technology studies is reported in the areas of radioactivity control, sodium systems, hydraulics and mechanics, and safety systems.

- 1760 (HEDL-TME--77-39) SODIUM ENGINEERING AND TECHNOLOGY. TECHNICAL PROGRESS REPORT, OCTOBER, NOVEMBER, DECEMBER 1977. Atwood, J.M. (comp.). (Hanford Engineering Development Lab., Richland, WA (USA)). May 1978. Contract EY-76-C-14-2170. 101p. AT.

This report represents a quarterly summary of progress made at Westinghouse Hanford Company in the areas of radioactivity control technology, sodium systems technology, hydraulics and mechanics, and safety systems development.

- 1761 (HEDL-TME--77-43) OPERATING EXPERIENCES AND ANALYSIS OF FFTF AND HTSF COLD TRAP ECONOMIZERS. Hassberger, J.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1977. Contract EY-76-C-14-2170. 73p. AT.

Operating experience with the Type-I and Type-III cold traps in the High-Temperature Sodium Facility demonstrated that the sodium regenerative heat exchangers (economizers) did not meet design specifications. Initial operating data from the Type-I economizer pointed to a major problem in tube-side flow maldistribution. Hydraulic tests on the Type-III economizer verified that the severity of flow maldistribution could be minimized through the use of tube orifices. Subsequent in-sodium heat transfer tests, however, showed that other significant problems existed, such as shell-side flow maldistribution and annulus free convection. The performances of the various economizers, both as-built and modified or projected, are described. Primary emphasis was placed on correcting tube-side deficiencies. Alternate mechanisms contributing to degraded performance are also discussed.

- 1762 (HEDL-TME--77-44) HIGH ^{240}Pu FTR/EMC EXPERIMENTS AND ANALYSIS: CARBIDE FUEL AND UO_2 BLANKET SUBASSEMBLY WORTHS. Ombrellaro, P.A.

(Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1977. Contract EY-76-C-14-2170. 40p. Dep. NTIS, PC A03/MF A01.

Carbide-plutonium fuel and UO_2 blanket subassembly worth measurements performed at ANL in the EMC/LWR were analyzed. Composition exchange worth calculations were performed for: (a) the replacement of high- ^{240}Pu fuel composition for low- ^{240}Pu fuel composition and carbide-plutonium fuel composition, successively, in the center subassembly of the core; (b) the replacement of low- ^{240}Pu fuel composition for carbide-plutonium fuel composition in one outer driver subassembly; and (c) the replacement of the radial reflector composition with UO_2 blanket composition in one subassembly of the radial reflector. The composition exchange worth calculations were performed in two-dimensional x,y geometry, using diffusion theory and perturbation theory. Each method produces about the same calculated-to-experimental bias factors.

- 1763 (HEDL-TME--77-47) CORE ENGINEERING TECHNICAL PROGRESS REPORT, APRIL--JUNE 1977. Peterson, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1977. Contract EY-76-C-14-2170. 64p. AT.

Progress in FFTF core development is presented for the following study areas: core physics, radiation and shield analysis, and core design.

- 1764 (HEDL-TME--77-48) CORE ENGINEERING TECHNICAL PROGRESS REPORT, JULY--SEPTEMBER 1977. Peterson, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 93p. AT.

Progress in FFTF core development is presented for the following study areas: core physics, radiation and shield analysis, and core design.

- 1765 (HEDL-TME--77-49) CORE ENGINEERING TECHNICAL PROGRESS REPORT, OCTOBER--DECEMBER 1977. Peterson, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1978. Contract EY-76-C-14-2170. 79p. Dep. NTIS, PC A05/MF A01.

Progress in FFTF core development is presented for the following study areas: (1) core physics and (2) radiation and shield analysis.

- 1766 (HEDL-TME--77-53) ENERGY 1A: A MODIFIED VERSION OF THE ENERGY 1 THERMAL-HYDRAULIC COMPUTER CODE FOR LMFBR WIRE WRAP SPACER FUEL ASSEMBLIES. Shearer, D.M. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1977. Contract EY-76-C-14-2170. 42p. AT.

ENERGY 1A is a modification of the ENERGY 1 thermal-hydraulic computer program developed at MIT. ENERGY 1A is useful for thermal-hydraulic sensitivity studies of large liquid metal fast breeder reactor fuel pin assemblies. The program is simple, easy to use, fast running, and readily accommodates large fuel pin assemblies with a minimum of input data. Calibration of ENERGY 1A is easily accomplished with experimental results or thermal-hydraulic results from a more rigorous computer program. For a flow solution given by a few input variables and input values for fuel pin energy generation ENERGY 1A predicts coolant and cladding midwall temperatures throughout the fuel pin assembly.

- 1767 (HEDL-TME--77-55) O/M STABILITY IN MIXED OXIDE FUEL PINS. Gottschalk, G. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1977. Contract EY-76-C-14-2170. 23p. Dep. NTIS, PC A02/MF A01.

Mixed Oxide fuel pellets have been analyzed for change in oxygen to metal ratio following extended storage in FFTF fuel pins.

- 1768 (HEDL-TME--77-58) BREEDER REACTOR REFERENCE CONTROL MATERIALS. SEMIANNUAL REPORT, JANUARY--JUNE 1977. Roake, W.E.; Claudson, T.T. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1977. Contract EY-76-C-14-2170. 61p. AT.

Progress on neutron absorber development activities from January 1977 through June 1977 is presented. Included in the report are descriptions of activities related to performance analysis, development testing, and fabrication technology.

- 1769 (HEDL-TME--77-59) CALCULATION OF PASSIVE SENSOR PERTURBATIONS IN FFTF. Rawlins, J.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1977. Contract EY-76-C-14-2170. 38p. AT.

The integral transport theory code RABBLE has been used to calculate resonance region broad-group capture and fission cross sections for ^{235}U , ^{238}U , ^{239}Pu , ^{240}Pu , and ^{241}Pu inside an inner driver FTR fuel pellet and inside a dosimeter container at two fuel temperatures, corresponding to FFTF operation at low power and at 400 MW. These resonance region cross sections are merged with the 42-group 300S set of cross sections processed through 1DX. The resulting composite 42-group cross section set is used to calculate fluxes and reaction rates in a pin cell model using the two dimensional S/sub n/ transport theory code DOT-IIW, using S-4 quadrature. Calculations are made for prototypical and nonprototypical conditions (temperature and material); perturbations to capture, fission, and other reaction rates are then calculated as a function of temperature and material conditions. The calculated perturbations are less than ± 4 percent for all the U and Pu isotopes considered. The calculations, therefore, indicate no overwhelming reason for performing the capture and fission rate measurements under any particular set of reactor conditions. Several reactions to be used in passive spectrum measurement are also considered, and some of the perturbations calculated for these reactions are larger than those of the fuel isotopes. It may be possible to measure a few spectral set reaction rate perturbations.

- 1770 (HEDL-TME--77-60) ANALYSIS OF MIXED OXIDE FUEL IRRADIATED IN EBR-II: MEASURED VS. PREDICTED BURNUP. Scott, L.D.; Dutt, D.S.; Wilson, D.R. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1978. Contract EY-76-C-14-2170. 19p. Dep. NTIS, PC A02/MF A01.

The calculation of burnup in mixed-oxide fuel pins irradiated in EBR-II is shown to agree with burnup measured by postirradiation radiochemical analysis. The mean percent deviation is 0.12% with a variance of 2.49%. This gives a high level of confidence that HIST, which uses 91% of the nominal EBR-II power, provides an accurate estimate of burnup.

- 1771 (HEDL-TME--77-61) CORROSION OF IRRADIATED STAINLESS STEEL IN HOT-TRAPPED SODIUM. Brehm, W.F.; Anantatmula, R.P.

(Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1977. Contract EY-76-C-14-2170. 40p. AT.

Corrosion and radionuclide release from irradiated stainless steel into hot-trapped sodium at 1120°F (604°C) were quantitatively determined. The oxygen content of the sodium was maintained at less than 0.01 parts per million. The results show that corrosion and radionuclide release were not reduced below values observed in tests at 0.5 and 2.5 parts per million oxygen in sodium. The significance of this result is that development of hot traps or soluble getters for purification system of sodium-cooled reactors is not warranted. 14 figures, 7 tables.

- 1772 (HEDL-TME--77-62) SELECTION OF PROCESS PARAMETERS FOR SODIUM REMOVAL VIA THE WATER VAPOR NITROGEN PROCESS. Crippen, M. (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1977. Contract EY-76-C-14-2170. 39p. AT.

This report reviews the water vapor nitrogen (WVN) process for removing sodium from stainless steel equipment, with emphasis on the basis for selecting final operating parameters. The process includes vapor treatment with 5% water at 160°F to 190°F, hot water rinsing, and hot gas drying.

- 1773 (HEDL-TME--77-63) REMOVAL OF SODIUM FROM THE IVHM ORIENTATION PLUG. Crippen, M.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1977. Contract EY-76-C-14-2170. 16p. AT.

A replica of the FFTF In-Vessel Handling Machine--Engineering Model (IVHM-EM) orientation plug was cleaned of sodium. It was demonstrated that cleaning time could be substantially reduced by either raising the moisture content of the water vapor-nitrogen gas or pressure cycling the reacting gas or both. The plug was repeatedly cleaned without damage.

- 1774 (HEDL-TME--77-66) FAST REACTOR SAFETY TECHNICAL PROGRESS REPORT, JANUARY--MARCH 1977. Simpson, D.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 162p. AT.

The report presented describes technical progress made during the reporting period, by Westinghouse-Hanford Company and supporting contractors, on fast reactor safety programs. These programs summarized include reactor safety activities supportive of the Fast Flux Test Facility and associated with development of base LMFBR technology.

- 1775 (HEDL-TME--77-67) FAST REACTOR SAFETY TECHNICAL PROGRESS REPORT, APRIL--JUNE 1977. Simpson, D.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1978. Contract EY-76-C-14-2170. 150p. Dep. NTIS (US Sales Only), PC A07/MF A01.

Progress is reported in the following studies related to FFTF and LMFBR safety: (1) reactor kinetics, (2) reactor safety, (3) system safety analysis, (4) safety development, (5) FSAR preparation and review, (6) safety test engineering, (7) fabrication development, and (8) safety systems development. (DG)

- 1776 (HEDL-TME--77-68) FAST REACTOR SAFETY. QUARTERLY TECHNICAL PROGRESS REPORT, JULY--SEPTEMBER 1977. Simpson, D.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1978. Contract EY-76-C-14-2170.

175p. AT.

The report describes technical progress on fast reactor safety programs. These programs include reactor safety activities (a) supportive of the Fast Flux Test Facility and (b) associated with development of base FBR technology.

- 1777 (HEDL-TME--77-70) IRRADIATED EBR-II DUCT IMPACT TESTS. Fish, R.L.; Steichen, J.M.; Duncan, D.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1977. Contract EY-76-C-14-2170. 47p. AT.

Experimental Breeder Reactor-II (EBR-II) duct components (annealed Type 304 SS) were impact-tested under loading and temperature conditions simulating fast reactor scrams. The ducts received a peak fluence of 9.8×10^{22} n/cm² (E greater than 0.1 MeV) during exposure in the EBR-II. The impact tests, performed on full duct components (45-in. long), used weights up to 13.6 kg (30 lb) which were dropped onto the upper end of the vertically-oriented components. These tests were conducted at 589°K and 811°K (600°F and 1000°F). To provide support stress-strain data for use in the analysis of duct impact test results, tensile tests were performed on specimens sectioned from a duct of equivalent fluence. Examinations of the duct components revealed no evidence of failure initiation or cracking due to the impact loads. Scanning electron fractography of tensile fracture surfaces showed no indication of low ductility "brittle" failure under the impact conditions simulated.

- 1778 (HEDL-TME--77-77) DIMENSIONAL ANALYSIS OF FFTF DRIVER FUEL ASSEMBLIES. Bunker, M.S.; Carlson, R.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1977. Contract EY-76-C-14-2170. vp. AT.

Dimensional analysis is performed on Fast Flux Test Facility (FFTF) driver fuel assemblies using a computer program called DUCT. The program takes dial indicator readings from a dimensional gage, models a hexagonal column in three dimensions; and compares the resultant model against acceptance criteria for twist, offset, and centerline alignment, or bow. The document describes the mathematical derivation of the DUCT calculations, the benchmark tests conducted prior to implementation, the techniques governing measurement, and the acceptance procedure.

- 1779 (HEDL-TME--77-78) THERMAL SHOCK CONSIDERATIONS PERTAINING TO INITIAL FTR CORE LOADING AND REFUELING. Lacko, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1977. Contract EY-76-C-14-2170. 33p. AT.

The results are presented of an analysis to determine the maximum allowable temperature differential between an FFTF fuel assembly and the reactor sodium pool that can be tolerated from the standpoint of thermal shock during initial core loading and subsequent refueling conditions. The analysis is based on a simplified and conservative one-dimensional thermal/inelastic stress model of an FFTF fuel assembly. The analysis shows that a fuel assembly must be preheated in the Core Component Conditioning Station (CCCS) to at least 280°F before loading into the reactor sodium pool, if the latter is assumed to be at 400°F refueling conditions.

- 1780 (HEDL-TME--77-79) EFFECT OF THERMAL CYCLING DURING IN-VESSEL STORAGE (IVS) OF FFTF

CORE COMPONENTS. Lacko, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1977. Contract EY-76-C-14-2170. 41p. AT.

The results are presented of an analysis that addresses the effects of thermal cycling of FFTF core components while stored in the In-Vessel Storage (IVS) location of the FTR. The analysis incorporates the latest IVS temperature fluctuations measured during the FFTF Outlet Feature Model Tests performed at HEDL. The results show that thermal fatigue surface cracking is possible on the surface of an FFTF core component handling socket. Linear elastic fracture mechanics theory was used to evaluate the crack propagation from an initial assumed crack depth. The maximum depth of crack penetration is not sufficient to cause failure of the handling socket during removal of a core component assembly from the FFTF reactor during refueling operations.

- 1781 (HEDL-TME--77-80) ALLOWABLE COOLANT OUTLET TEMPERATURE DIFFERENTIALS FOR FTR TEST ASSEMBLIES IN DRIVER POSITIONS. Lacko, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1977. Contract EY-76-C-14-2170. 20p. AT.

Results are reported of an analysis to establish allowable coolant outlet temperature differentials for driver position test assemblies relative to their surrounding assemblies in the Fast Test Reactor (FTR). The allowable coolant temperature differentials for driver position test assemblies supplement those previously established for an Open Test Assembly (OTA) and the surrounding assemblies. Both lower and upper bounds of a driver position test assembly coolant temperature differential relative to surrounding assemblies have been established for a range of FFTF core inlet coolant temperatures between 600°F and 680°F. The fuel assembly duct tube was found to be the critical structural member that limits the allowable sodium coolant temperature differential.

- 1782 (HEDL-TME--77-83) EX-REACTOR TRANSIENT FISSION GAS RELEASE STUDIES, FUEL PIN PNL-2-4. Hinman, C.A.; Slagle, D.D. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1978. Contract EY-76-C-14-2170. 258p. AT.

Fast neutron irradiated UO₂--PuO₂ was rapidly heated at approximately 200°C/sec in a laboratory furnace. Transient gas release and fuel physical behavior were monitored during the rapid heating. Extensive microstructural examinations were conducted on the fuel which had been heated to various peak temperatures (less than melting). It was found that fission gas release behavior from PNL-2 fuel closely resembled earlier measured release rates from PNL-5 fuel. A gross-swelling temperature threshold exists for PNL-2 fuel between 2400°C and 2700°C. Neither grain boundary shattering nor thermal gradient induced diffusion of gas bubbles was observed. However, a high degree of nonequilibrium of gas bubble pressure and fuel surface tension was encountered. The evidence gathered in this work can be explained by a stress gradient directed forced diffusion mechanism.

- 1783 (HEDL-TME--77-85) PROGRESS IN RADIONUCLIDE TRAP DEVELOPMENT. McGuire, J.C.; Brehm, W.F. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1977. Contract EY-76-C-14-2170. 28p. AT.

The report describes a radionuclide trap being developed to remove gamma-emitting

irradiation products from primary sodium in a liquid metal breeder reactor. The radionuclides, principally ^{54}Mn and ^{60}Co , are removed from sodium in the core area of the reactor to prevent their deposition in the heat exchanger. Prototype traps have been evaluated for the past three years and a full-scale version is now in test in EBR-II.

- 1784 (HEDL-TME--77-86) CALIBRATION OF A FUEL-TO-CLADDING GAP CONDUCTANCE MODEL FOR FAST REACTOR FUEL PINS. Baker, R.B. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1978. Contract EY-76-C-14-2170. 132p. Dep. NTIS, PC A07/MF A01.

The report presents refined methods for calculation of fuel temperatures in $\text{PuO}_2\text{-UO}_2$ fuel in Fast Breeder Reactor (FBR) fuel pins. Of primary concern is the calculation of the temperature changes across the fuel-to-cladding gap of pins with fuel burnups that range from 60 to 10,900 Mwd/MTM (0.006 to 1.12 at.%). Described in particular are: (1) a proposed set of heat transfer formulations and corresponding material properties for modeling radial heat transfer through the fuel and cladding; and (2) the calibration of a fuel-to-cladding gap conductance model, as part of a thermal performance computer code (SIEX-M1) which incorporates the proposed heat transfer expressions, using integral thermal performance data from two unique in-reactor experiments. The test data used are from the HEDL P-19 and P-20 experiments which were irradiated in the Experimental Breeder Reactor Number Two (EBR-II), for the Hanford Engineering Development Laboratory (HEDL).

- 1785 (HEDL-TME--77-89) SWAB TEST: A METHOD TO DETERMINE SURFACE CONTAMINATION OF STAINLESS STEEL REACTOR COMPONENTS BY CHLORIDE AND FLUORIDE. Stromatt, R.W.; Lambert, M.C. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1977. Contract EY-76-C-14-2170. 35p. AT.

A method--the swab test--to determine the surface contamination of stainless steel components of the Fast Flux Test Facility by chloride and fluoride is described. The Auger electron spectroscopy and analytical chemistry experiments to establish credibility for the procedure are presented.

- 1786 (HEDL-TME--77-91) THREE-DIMENSIONAL INELASTIC ANALYSIS OF A SMALL LIQUID SODIUM VALVE. Weiner, E.O. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1978. Contract EY-76-C-14-2170. 198p. AT.

Portions of document are illegible.

A three-dimensional inelastic analysis of a small liquid sodium valve for the Fast Flux Test Facility was conducted to assess ratcheting and creep damage at temperatures to 1200°F using the MARC-CDC finite element program. A single mesh of twenty-node solid elements, representing the Y-type cylinder intersection is used for both the heat transfer and mechanical computer runs. Material behavior is governed by kinematic and creep strain hardening, and the loading consists of steady mechanical loads with cyclic fluid thermal transients. The resulting creep damage is significantly smaller than that of a corresponding axisymmetric analysis, and plastic strains are confined to a single region.

- 1787 (HEDL-TME--77-94) HEDL MIXED OXIDE FUELS FABRICATION DEVELOPMENT. QUARTERLY PROGRESS REPORT, JANUARY--MARCH 1977.

Bardsley, R.E. (comp.). (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1977. Contract EY-76-C-14-2170. 25p. Dep. NTIS, PC A02/MF A01.

Progress in LMFBR fuel fabrication development is described under the following headings: fuel characterization and attribute engineering, test fuel and pin fabrication, ANL/SLSF and IPE safety development pin fabrication, tag gas capsule fabrication, fuel assembly fabrication, and FFTF test articles.

- 1788 (HEDL-TME--77-95) LMFBR MIXED OXIDE FUELS FABRICATION DEVELOPMENT SEMIANNUAL, JANUARY--JUNE 1977. Bardsley, R.E. (comp.). (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1977. Contract EY-76-C-14-2170. 30p. AT.

Work is reported on the following: fuel characterization and attribute engineering, high-performance oxide fabrication studies, ANL/SLSF and IPE safety development fuel pin fabrication, test fuel and pin fabrication, tag gas capsule fabrication, fuel assembly fabrication development, fuel assembly, and FFTF test articles. (DLC)

- 1789 (HEDL-TME--77-96) COUPLE M: A COMPUTER PROGRAM WHICH MODELS THERMAL-HYDRAULIC AND DIMENSIONAL CHANGE BEHAVIOR OF A MULTI-PIN FBR FUEL ASSEMBLY. Shearer, D.M. (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1977. Contract EY-76-C-14-2170. 58p. AT.

The COUPLE M computer program is presented. COUPLE M models both the thermal-hydraulic and dimensional change behavior of Fast Breeder Reactor fuel assembly during irradiation. The program is described and input instructions and variables are given. A program listing and sample problem are provided in the appendix.

- 1790 (HEDL-TME--77-98) MGO INSULATED COAXIAL INSTRUMENT CABLE DEVELOPMENT FOR FFTF IN-VESSEL TEMPERATURE AND VOLTAGE CONDITIONS. Chan, A.I.Y.; Cannon, C.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1978. Contract EY-76-C-14-2170. 33p. Dep. NTIS, PC A03/MF A01.

MgO insulated nuclear instrument cable designed for use in the Fast Flux Test Facility or breeder reactor in-vessel service environments were successfully tested to temperatures to 650°C. Breakdown Pulse Noise (BPN) thresholds in excess of 800 Vdc at 650°C were achieved on the bulk cable; cable resistivities at 650°C were 10⁶ ft. The effect of 1000°C annealing was a minimal reduction of cable resistivity with no observed effect on BPN threshold. No differences in insulation resistance and BPN thresholds were detected between cables containing copper electrical conductors and cables containing Inconel 600 conductors. Thus a copper conductor cable type was established to be suitable for both signal transmission and high voltage bias applications. The tested bulk cables exceeded final cable development program objectives: no BPN at 650°C, 800 Vdc continuous, and 10⁶ ft resistivity at 550°C in copper conductor MgO insulated bulk cables.

- 1791 (HEDL-TME--77-99) INTERMEDIATE SCALE SODIUM-CONCRETE REACTION TESTS. Hassberger, J.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1978. Contract EY-76-C-14-2170. 160p. AT.

Twelve intermediate scale, sodium-concrete reaction tests have been completed. These tests involve spilling hot (1250 to 1600°F, 677 to

871°C) sodium onto 1 ft³ (0.028 m³) specimens of magnetite or limestone concretes. The resulting reactions were allowed to continue for from 2 to 24 hours, at which time the sodium was decanted from the test article. Included are discussions of the observed penetration depths, chemical analyses of the reaction products, characterization of the evolved gasses and a brief summary of laboratory scale experiments performed earlier.

- 1792 (HEDL-TME--77-100) EXPERIENCE IN THE ANALYSIS OF NUCLEAR-GRADE BORON CARBIDE. Stromatt, R.W.; Delvin, W.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1978. Contract EY-76-C-14-2170. 197p. Dep. NTIS (US Sales Only), PC A09/MF A01.

A review of the analytical chemistry methods used at the Hanford Engineering Development Laboratory (HEDL) for the analysis of boron carbide and a summary of results obtained from the analysis of the boron carbide during fabrication of control rods for the Fast Flux Test Facility (FFTF) are presented. A brief review of the qualification and control program that governed the analytical chemistry laboratories during production of pellets for use with FFTF cores I and II is included. Detailed HEDL analytical chemistry methods are given in the Appendix.

- 1793 (HEDL-TME--77-102) HIGH TEMPERATURE LINER FEATURE TEST, FT-3. Simmons, L.D.; Wierman, R.W.; Jensen, R.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1978. Contract EY-76-C-14-2170. 76p. AT.

The High Temperature Liner Feature Test, FT-3, was designed to test the adequacy of one wall design for breeder reactor cavity or equipment cell liners to sustain the rapid heating from an accidental liquid metal coolant spill without failure or leakage. A segment of a cylindrical wall, with steel liner plate welded to steel structure embedded in the structural concrete, was heated rapidly by radiant heaters. The liner plate buckled early in the test so that stresses in the liner were limited to the magnitude required to cause buckling, and no liner or structural failure occurred. The rapid heating caused extensive shallow fracturing in the concrete directly behind the liner plate, but this was judged to have no adverse consequences and did not affect the integrity of the liner.

- 1794 (HEDL-TME--77-105) EVALUATION OF ROD DROP STABILITY TESTS IN FFTF. Harris, R.A.; Sevenich, R.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1978. Contract EY-76-C-14-2170. 55p. AT.

An evaluation of the adequacy of rod drop-type experiments to verify the stability margin in the FFTF is presented. This evaluation is based on simulations using models with the expected characteristics of the reactor. The assumed driving function for the experiments is the scrambling of a nearly fully inserted secondary control rod. It is shown that an adequate stability margin can be assured from the inferred Nyquist diagrams of the open-loop transfer function. The experiments cannot resolve the magnitude of the individual feedback mechanisms; therefore, predictions of the stability margins at reactor states other than the measurement state must rely on extrapolations of feedback mechanism magnitudes estimated from other corollary experiments or the wide stability margin expected in the FFTF. Positioning of the drop rod will result in a

major perturbation on the spatial power density in the reactor and thus will have a significant impact on nearby irradiation test subassemblies during the actual testing period.

- 1795 (HEDL-TME--77-107) DETECTION OF FEEDBACK RESONANCES IN FFTF USING ROD DROP STABILITY TESTS. Harris, R.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1978. Contract EY-76-C-14-2170. 24p. AT.

An analysis is presented of simulated rod drop experiments in the Fast Flux Test Facility using reactor models with unanticipated resonant behavior. The analysis shows that a narrow resonance in the reactor transfer function should be detected if it resulted in an increase in the frequency response function gain of 26% over the gains at nearby frequencies. This detection level assumes no prior knowledge of the resonance and no attempt to model it in the analysis. Changes in the total reactor frequency response function gain of approximately 13% are shown to be detectable if consistent experimental data before and after the change are available.

- 1796 (HEDL-TME--77-108) REACTIVITY SURVEILLANCE PROCEDURES FOR ANOMALY DETECTION. Harris, R.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1978. Contract EY-76-C-14-2170. 76p. AT.

The procedures for detecting unexpected reactivity changes in the FFTF are presented. These procedures are implemented on the Digital Data Handling and Display System (DDH and DS) and thus provide rapid indication of abnormal conditions within the core. The algorithms which use the thermal-hydraulic, the control rod position and the neutron flux sensor information to predict the reactivity state of the core are described and are shown to be adequate for the currently planned mode of operation of the reactor. Sufficient flexibility exists to revise these algorithms to reflect observed reactivity state characteristics by using calibration procedures. These procedures are also discussed. The expected uncertainties in the procedures due to uncertainties in both the sensors and algorithms are presented. These show that for acceptable false alarm rates, the alarm limit must be set fairly high (approximately 20 cents after calibration of the procedures) during major power changes. The limit may ultimately be reduced to approximately ± 2 cents for steady power operation.

- 1797 (HEDL-TME--77-109) BREEDER REACTOR REFERENCE CONTROL MATERIALS. SEMIANNUAL REPORT, JULY--DECEMBER 1977. Roake, W.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1978. Contract EY-76-C-14-2170. 43p. AT.

Progress on neutron absorber development activities from July 1977 through December 1977 is presented. Included in the report are descriptions of activities related to performance analyses, development testing and fabrication technology.

- 1798 (HEDL-TME--78-1) LARGE HETEROGENEOUS REFERENCE FUEL DESIGN STUDY. FINAL REPORT. Chandler, J.C.; Marr, D.R.; McCurry, D.C.; Cantley, D.A.; Omberg, R.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1978. Contract EY-76-C-14-2170. 121p. Dep. NTIS, PC A06/MF A01.

The fuel pin geometry used in the Fast Test

Reactor (FTR) and considered for use in a prototypic small breeder installation has been evaluated through a comprehensive development program during recent years so that an extensive data base exists for extrapolation of the FTR fuel pin and geometry to larger installations. The objectives of the Large Heterogeneous Reference Fuel Design Study (LHRFDS) were to evaluate the performance capabilities of the reference FTR fuel pin in a 1200 MWe reactor plant installation and to identify a corresponding optimum core design. For purposes of the study, a reference fuel assembly was defined, corresponding to the FTR fuel pin with axial blankets placed in a 217-pin assembly. The reference assembly was used to define a reference core, which approximated FTR technology extrapolated to prototypic small breeder conditions. Performance indices of the reference core were provided by ARD during the study.

- 1799 (HEDL-TME--78-10) FFTF VIBRATION OPEN TEST ASSEMBLY (VOTA) ENGINEERING MODEL CALIBRATION PROGRAM. White, R.G.; Simmons, M.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1978. 70p. AT. Natural frequencies, mode shapes, and damping values were experimentally determined for an engineering model of the Fast Flux Test Facility (FFTF) vibration open test assembly (VOTA), and optimum accelerometer locations were determined for the VOTA plant unit. Testing was conducted at ambient temperature in air and quiescent water. The VOTA was installed in a closed-loop in-reactor assembly (CLIRA) spoolpiece and a prototypic inlet nozzle receptacle and liner. During the application of random excitation, dynamic response data were acquired for each of a variety of constraints applied to the VOTA top load pad and above-core load pad.

- 1800 (HEDL-TME--78-11) TIME STEP MOTA THERMOSTAT SIMULATION. Guthrie, G.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1978. Contract EY-76-C-14-2170. 29p. Dep. NTIS, PC A03/MF A01.

The report details the logic, program layout, and operating procedures for the time-step MOTA (Materials Open Test Assembly) thermostat simulation program known as GYRD. It will enable prospective users to understand the operation of the program, run it, and interpret the results. The time-step simulation analysis was the approach chosen to determine the maximum value gain that could be used to minimize steady temperature offset without risking undamped thermal oscillations. The advantage of the GYRD program is that it directly shows hunting, ringing phenomenon, and similar events. Programs BITT and CYLB are faster, but do not directly show ringing time.

- 1801 (HEDL-TME--78-12) SMALL THERMAL OSCILLATION ANALYSIS OF THE MOTA. Guthrie, G.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1978. Contract EY-76-C-14-2170. 57p. Dep. NTIS, PC A04/MF A01.

The MOTA (Materials Open Test Assembly) was designed to achieve a degree of thermal regulation compatible with the generation of useful materials property data obtained by irradiation of candidate reactor structural materials in the FFTF. Attaining a high degree of regulation is limited by the necessity to avoid undamped thermal oscillations. The report documents some of the analyses used to select useable configurations and determine effects of

parameter choices, and investigates limitations on allowable gains of the sensor-control-valve assembly. The main purpose of the document is to make the methods available to others, rather than to give a tabulation of specific numerical results.

- 1802 (HEDL-TME--78-13) MECHANICAL BEHAVIOR OF FAST REACTOR FUEL PIN CLADDING SUBJECTED TO SIMULATED OVERPOWER TRANSIENTS. Johnson, G.D.; Hunter, C.W. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1978. Contract EY-76-C-14-2170. 195p. Dep. NTIS, PC A09/MF A01.

Cladding mechanical property data for analysis and prediction of fuel pin transient behavior were obtained under experimental conditions in which the temperature ramps of reactor transients were simulated. All cladding specimens were 20% CW Type 316 stainless steel and were cut from EBR-II irradiated fuel pins. It was determined that irradiation degraded the cladding ductility and failure strength. Specimens that had been adjacent to the fuel exhibited the poorest properties. Correlations were developed to describe the effect of neutron fluence on the mechanical behavior of the cladding. Metallographic examinations were conducted to characterize the failure mode and to establish the nature of internal and external surface corrosion. Various mechanisms for the fuel adjacency effect were examined and results for helium concentration profiles were presented. Results from the simulated transient tests were compared with TREAT test results.

- 1803 (HEDL-TME--78-14) GAS TAG LOCATION SYSTEMS FOR FFTF AND CRBRP FAILED ASSEMBLIES. Figg, J.A.; McCormick, N.J.; Schenter, R.E.; Schmittroth, F.; DeClue, J.F.; Holt, F.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1978. Contract EY-76-C-14-2170. 136p. AT.

The report is directed at the computational aspects associated with the gas tag location system. The analytical techniques used to select the isotopic composition of the gases and the computer program to prescribe the blending of the gas mixtures during fabrication are described, and the computer program for identifying a failed assembly using the measured ratios is discussed. The tag compositions for the FFTF and CRBRP reactors also are presented along with sample calculations for the identification of failed assemblies using postulated tag gas releases.

- 1804 (HEDL-TME--78-18) FFTF PREOPERATIONAL ENVIRONMENTAL SURVEY PROGRAM. Horning, T.W. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1978. Contract EY-76-C-14-2170. 35p. AT.

The Preoperational Environmental Survey Program establishes a pre-startup (initial criticality) background data base which is essential for comparative evaluation of routine monitoring information. Type, frequency, and location of specific surveys are consistent with those required by the Environmental Technical Specifications for the Fast Flux Test Facility (FFTF FSAR Chapter 18). The objective of the radiological monitoring program is to quantify concentrations of radioactive materials and direct radiation levels in the environs in order to verify that as a result of plant operations, persons in unrestricted areas do not receive radiation doses greater than those specified in 10 CFR 50 Appendix I. In addition, measurements will provide assurance that off-site doses resulting from radioactive materials in plant effluents are as low as

practicable. These goals should be accomplished by the environmental monitoring program outlined. It is emphasized that the FFTF is one of several facilities on the Hanford site. The environmental impact to unrestricted areas attributable to FFTF operations will be evaluated as part of the overall determination of the impact due to all Hanford facilities. However, only programs which are directly applicable to measurement of the environmental effects of the FFTF are discussed as part of the FFTF Routine Environmental Monitoring Program.

- 1805 (HEDL-TME--78-20) SODIUM ENGINEERING AND TECHNOLOGY TECHNICAL PROGRESS REPORT, JANUARY, FEBRUARY, MARCH 1978. Atwood, J.M. (comp.). (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1978. Contract EY-76-C-14-2170. 102p. AT.

The report presents a quarterly summary of progress made at Westinghouse Hanford Company in the areas of radioactivity control technology, sodium systems technology, hydraulics and mechanics, and safety systems development.

- 1806 (HEDL-TME--78-21) SODIUM ENGINEERING AND TECHNOLOGY. TECHNICAL PROGRESS REPORT, APRIL, MAY, JUNE 1978. Atwood, J.M. (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1978. Contract EY-76-C-14-2170. 147p. AT.

Progress in LMFR-related research is reported under the following headings: radioactivity control technology, sodium systems technology, safety systems development, hydraulics, and mechanics.

- 1807 (HEDL-TME--78-21) SODIUM ENGINEERING AND TECHNOLOGY. TECHNICAL PROGRESS REPORT, APRIL, MAY, JUNE 1978. Atwood, J.M. (comp.). (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1978. Contract EY-76-C-14-2170. 146p. AT.

This report presents a quarterly summary of progress made at Westinghouse Hanford Company in the areas of radioactivity control technology, sodium systems technology, safety systems development, and hydraulics and mechanics.

- 1808 (HEDL-TME--78-23) SODIUM VAPOR DEPOSIT TECHNOLOGY. Funk, C.W. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1979. Contract EY-76-C-14-2170. 93p. AT.

This report reviews available technology of sodium vapor deposits in sodium cooled reactors; describes operational problems attributed to such deposits; and discusses processes for deposit formation, removal, and prevention. The document presents data concerning the amount, appearance, condition, and effect of deposits on performance of Fast Flux Test Facility (FFTF) components exposed to high-temperature sodium vapor during testing. Components examined include various reactor plugs, the in-vessel handling machine, and the instrument tree.

- 1809 (HEDL-TME--78-28) OPERATIONAL EXPERIENCE WITH A MODIFIED LIQUID METAL OXYGEN METER. Chulos, L.E.; Rake, R.R. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1978. Contract EY-76-C-14-2170. 19p. AT.

A commercially available air reference electrode liquid metal oxygen activity meter was modified for use in a flowing sodium

system. The unmodified air reference electrode meter requires a layer of vapor-deposited platinum on the ceramic electrolyte to complete electrical contact. Failure of this meter would release platinum to the loop sodium, which would interfere with analysis of mass transfer deposits. The modified meter eliminates the platinum and exhibits better conductivity for faster response to changes in oxygen concentration. It is also more rugged, which contributes to longer meter life.

- 1810 (HEDL-TME--78-35) COMPARISON OF CONTAINMENT SYSTEMS FOR LARGE SODIUM-COOLED BREEDER REACTORS. Seeman, S.E.; Armstrong, G.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1978. Contract EY-76-C-14-2170. 199p. Dep. NTIS, PC A09/MF A01.

Three types of containment configurations for large breeder reactors (1000 MWe/2500 MWt) were compared to determine characteristic responses of each type to selected hypothetical accident conditions. The three types of system configurations were the single, double and containment-confinement designs. Two types of postulated hypothetical accidents were applied as a basis for evaluation--the energetic core disassembly accident and the whole core melt-through accident. More than 20 separate cases were analyzed in which design and accident assumptions were varied. Based on this study the containment-confinement configuration is shown to be the most promising for limiting off-site radiation doses for large sodium-cooled breeder reactors given the specific assumptions on the initial accident conditions.

- 1811 (HEDL-TME--78-36) FRICTION, WEAR, CORROSION, AND SELF-WELDING OF TRIBALOG 700 HARDSURFACING IN SODIUM. Johnson, R.N.; Farwick, D.G. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1978. Contract EY-76-C-14-2170. 83p. AT.

Tribalog 700 is a nickel-base hardsurfacing alloy that exhibits low friction, excellent wear resistance, and low corrosion rates in sodium; as such, it is well suited for use in high temperature liquid metal environments. Its freedom from cobalt makes it particularly attractive for nuclear applications. Tribalog 700, rubbing against itself in sodium, exhibits among the lowest friction coefficients measured for metallic materials at high temperatures. In thermal cycling tests in sodium, detonation-gun coatings were unaffected by 120 cycles while all plasma coatings, except a special duplex coating, cracked or spalled within 30 cycles. Tribalog 700 showed good resistance to self-welding at temperatures up to 540°C (1000°F) in sodium and at temperatures up to 315°C (600°F) in argon and sodium vapor. The physical properties of Tribalog 700 are also described.

- 1812 (HEDL-TME--78-38) NATIONAL FRICTION, WEAR, AND SELF-WELDING PROGRAM STATUS AND PLANS. Johnson, R.N. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1978. Contract EY-76-C-14-2170. 95p. AT.

This report documents the status and plans of the National Friction, Wear, and Self-Welding Program. The primary objective of this program is to qualify materials for tribological applications through selection, development, and testing activities. The long-term goal is to develop a base list of well-characterized, wear-resistant materials, practical and economical fabrication process, and qualified sources of supply. The report reviews the status of testing and development activities, identifies program needs, and

recommends courses of action and priorities for future testing and development.

- 1813 (HEDL-TME--78-38) NATIONAL FRICTION, WEAR, AND SELF-WELDING PROGRAM STATUS AND PLANS. Johnson, R.N. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1978. Contract EY-76-C-14-2170. 92p. AT.

The report documents the status and plans of the National Friction, Wear, and Self-Welding Program. The primary objective of this program is to qualify materials for tribological applications through selection, development, and testing activities. The long-term goal is to develop a base list of well-characterized, wear-resistant materials, practical and economical fabrication process, and qualified sources of supply. The report reviews the status of testing and development activities, identifies program needs, and recommends courses of action and priorities for future testing and development.

- 1814 (HEDL-TME--78-39) SUBCRITICAL REACTIVITY MONITORING: NEUTRON YIELDS FROM SPONTANEOUS (α, n) REACTIONS IN FFTF FUEL. Ombrellaro, P.A.; Johnson, D.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1978. Contract EY-76-C-14-2170. 37p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Neutron yields resulting from the reaction of alpha particles emitted by heavy metal isotopes with isotopes of oxygen are evaluated. The yields are evaluated for various alpha-decaying isotopes that are initially present or built up during irradiation in FFTF fuels. Calculations are performed using a sum of probabilities for $^{17}\text{O}(\alpha, n)^{20}\text{Ne}$ and $^{18}\text{O}(\alpha, n)^{21}\text{Ne}$ reactions in fuels which have been evaluated using basic nuclear data. The calculated yields are in agreement with those found in the literature except for the ^{242}Cm and ^{244}Cm isotopes of curium where the discrepancy is approximately one order of magnitude. The curium isotopes are present only in irradiated fuel and build up with increased irradiation. However, for ^{244}Cm the predominant contribution of neutrons to the total neutron source results from spontaneous fission rather than the (α, n) reactions of the alpha particles emitted by ^{244}Cm with oxygen. For ^{242}Cm , the discrepancy is more significant since previous evaluations of the (α, n) yield showed it to be comparable to the spontaneous fission yield. The method used for the calculation of the (α, n) reaction yields provide the most accurate neutron yield evaluation currently available with existing nuclear data.

- 1815 (HEDL-TME--78-42) CORE ENGINEERING TECHNICAL PROGRESS REPORT, JANUARY, FEBRUARY, MARCH 1978. Peterson, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1978. Contract EY-76-C-14-2170. 116p. Dep. NTIS, PC A06/MF A01.

Progress in FFTF core development is summarized for the following study areas: reactor kinetics, radiation and shield analysis, and core design. (DG)

- 1816 (HEDL-TME--78-43) CORE ENGINEERING TECHNICAL PROGRESS REPORT, APRIL, MAY, JUNE 1978. Peterson, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1978. Contract EY-76-C-14-2170. 102p. AT.

Progress is summarized in the following FFTF core development areas: core physics, radiation and shield analysis, and core design. (DG)

- 1817 (HEDL-TME--78-46) SODIUM FLOW DISTRIBUTION IN TEST FUEL ASSEMBLY P-23B. Taylor, J.P.S. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1978. Contract EY-76-C-14-2170. 83p. Dep. NTIS, PC A05/MF A01.

Relatively large cladding diametral increases in the exterior fuel pins of HEDL's test fuel subassembly P-23B were successfully explained by a thermal-hydraulic/solid mechanics analysis. This analysis indicates that while at power, the subassembly flow was less than planned and that the fuel pins were considerably displaced and bowed from their nominal position. In accomplishing this analysis, a method was developed to estimate the sodium flow distribution and pin distortions in a fuel subassembly at power.

- 1818 (HEDL-TME--78-51) DUCTILITY AND STRENGTH OF FFTF/CRBRP STRUCTURAL MATERIALS IRRADIATED IN VARIOUS SPECTRA. INTERIM REPORT. Ward, A.L.; Blackburn, L.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1978. Contract EY-76-C-14-2170. 76p. AT.

Tension test results have been obtained for the initial half of the FFTF/CRBRP Structural Materials Irradiation Experiment which was designed to investigate in EBR-II the relatively low temperature (659 to 676°K) effects of neutron fluence (to $6.4 \times 10^{22} \text{ n/cm}^2$) interpretation of behavioral and spectrum (average neutron energies from 0.19 to 0.77 MeV) on the strength and ductility of 304 SS, 316 SS, weld-deposited 308L SS and Alloy 718. An interim (fluence of $1.9 \times 10^{22} \text{ n/cm}^2$) interpretation of behavioral trends in these materials as they relate to fluence limits for out-of-core FBR component applications indicates that, by comparison, the energy-independent fluence limits for 0.10 residual tensile elongation currently in use for CRBRP are conservative for the stainless steel materials.

- 1819 (HEDL-TME--78-52) ASSESSMENT OF THE HEDL P-23 SPECIAL PINS TEST. METHODS TO CONTROL FUEL-CLADDING CHEMICAL INTERACTION. Wilson, C.W. (Hanford Engineering Development Lab., Richland, WA (USA)). Oct 1978. Contract EY-76-C-14-2170. 130p. AT.

An irradiation experiment was conducted in Experimental Breeder Reactor-II (EBR-II) to evaluate placement of chromium, niobium, titanium, and vanadium into Liquid Metal Fast Breeder Reactor (LMFBR) fuel pins for the purpose of controlling fuel-cladding chemical interaction (FCCI). Effective FCCI inhibition was dependent on adequate axial dispersal of absorber materials and was achieved with Nb and V fuel coatings. Nondestructive examination data from 21 fuel pins and destructive examination data from 11 fuel pins are assessed.

- 1820 (HEDL-TME--78-57) USER'S MANUAL HEATUP CLOSED CELL H AND V FAILURE TRANSIENT ANALYSIS CODE. Chandler, J.C. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1978. Contract EY-76-C-14-2170. 128p. Dep. NTIS, MF A01.

Portions of document are illegible. HEATUP, a closed cell cooling system failure transient analysis code for Liquid Metal Fast Breeder Reactor (LMFBR) licensing analyses is described. The code replaces a modified version of SOFIRE-II formerly used for cooling failure analyses on the Fast Flux Test Facility (FFTF). HEATUP calculates heat transfer rates within

the cell structure and produces temperature and pressure histories based on input-controlled process sodium conditions. The output is designed to be readily usable for emergency load sequencing and structural analyses.

- 1821 (HEDL-TME--78-66) ANALYSIS OF FUEL-CLADDING CHEMICAL INTERACTION IN MIXED-OXIDE FUEL AT HIGH BURNUP. Lawrence, L.A.; Weber, J.W.; Devary, J.L. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1978. Contract EY-76-C-14-2170. 52p. Dep. NTIS, PC A04/MF A01.

The character and extent of fuel-cladding chemical interaction (FCGI) were established for mixed uranium-plutonium oxide fuels at an initial oxygen-to-metal ratio of 1.984 irradiated in EBR-II to a peak burnup of approximately 90.0 MWd/kgM. Depths of interaction were correlated with fuel pin operating conditions, i.e., burnup and temperature, and results discussed in terms of a change in reaction kinetics.

- 1822 (HEDL-TME--78-69) BREEDER REACTOR REFERENCE CONTROL MATERIALS. SEMIANNUAL STATUS NEWSLETTER, JANUARY--JUNE 1978. Roake, W.E. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1978. Contract EY-76-C-14-2170. 70p. AT.

Progress on neutron absorber development activities from January 1978 through June 1978 is presented. Included in the report are descriptions of activities related to performance analyses, development testing and fabrication technology.

- 1823 (HEDL-TME--78-78) GAS COOLING TEST WITH A FULL-SCALE 217-PIN BUNDLE ASSEMBLY. Boasso, C.J.; Madson, A.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Nov 1978. Contract EY-76-C-14-2170. 203p. AT.

In forced-convection argon gas, the heat-transfer characteristics of a modified, electrically-heated, 217-pin Fast Flux Test Facility (FFTF) fuel bundle are investigated. Test data is presented to supplement analytical productions of heat transfer in a forced argon convection mode for expended FFTF and Clinch River Breeder Reactor fuel assemblies. The test assembly is described in detail, with attention focused on the 217-pin electrically-heated fuel bundle. Test variables, including assembly power, argon flow rate, and fuel assembly pressure drop are presented and discussed.

- 1824 (HEDL-TME--78-80) EXPERIMENTAL STUDY OF HYDROGEN JET IGNITION AND JET EXTINGUISHMENT. Wierman, R.W. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1979. Contract EY-76-C-14-2170. 114p. Dep. NTIS, PC A06/MF A01.

Two phases are described of an experimental study that investigated: (1) the ignition characteristics of hydrogen--sodium jets, (2) the formation of hydrogen in sodium--humid air atmospheres, and (3) the extinguishment characteristics of burning hydrogen--sodium jets. Test conditions were similar to those postulated for highly-improbable breeder reactor core melt-through accidents and included: jet temperature, jet velocity, jet hydrogen concentration, jet sodium concentration, atmospheric oxygen concentration, and atmospheric water vapor concentration.

- CAPTURE OF CONDENSIBLE AIRBORNE CONTAMINANTS IN AN AQUEOUS SCRUBBING SYSTEM. Postma, A.K.; Hilliard, R.K. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1978. Contract EY-76-C-14-2170. 84p. Dep. NTIS, PC A05/MF A01.

The fate of condensible contaminants in an aqueous scrubbing system was evaluated. Knowledge of the behavior of volatile fission product compounds is important in evaluating the effectiveness of emergency air cleaning systems proposed for use in containment systems of breeder reactor plants. When a high temperature air stream passes through a spray quench chamber, very large cooling rates occur in the drop boundary layers. These large cooling rates cause large supersaturations in airborne concentrations of condensible contaminants, and one predicts that most condensation would take place through homogeneous nucleation. The very small particles formed would agglomerate, and attach to sodium aerosol particles which would be present. In the study the overall removal efficiency of volatile fission product species (typified by NaI, SeO₂, and Sb₂O₃) in an air cleaning train (quench chamber, venturi scrubber, and fibrous bed) was theoretically evaluated. The overall removal efficiency of condensible materials was found to be lower than that for sodium compound aerosols because the freshly condensed particles would be smaller in size. For a base case, a removal efficiency of 99.97 percent was predicted for condensible materials. The fibrous bed scrubber exhibited superior particle removal characteristics for small particles compared to the quench chamber and venturi scrubber. Its removal efficiency exceeded 97 percent for even the most penetrating particle size (about 0.4 micron aerodynamic diameter). Therefore, all condensible fission products would be removed with efficiencies exceeding 97 percent.

- 1826 (HEDL-TME--78-84) EBR-II SPECTRAL PARAMETERS, RUN 75D. Long, C.L.; Uiseth, J.A.; Lippincott, E.P.; Graves, N.J. (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1979. Contract EY-76-C-14-2170. 77p. Dep. NTIS, PC A05/MF A01.

A dosimetry flux mapping experiment was performed in the EBR-II reactor during Run 75D. Fission and reaction rate data were analyzed from 54 flux spectral sets, containing up to 18 dosimeters and numerous interspersed Fe gradient wires, located in six subassemblies from Row 2 out through Row 8. The dosimeters were located axially from 70 cm below to 75 cm above reactor midplane. The fission and reaction rate data were used to generate neutron spectra from which r, z matrices of nuclear parameters were generated. The report includes parameter matrices for total flux, flux with $E > 0.1$ MeV, neutron average energy (E_{bar}), and atom displacement cross section values for stainless steel.

- 1827 (HEDL-TME--78-89) SPENT THERMAL REACTOR FUEL ASSEMBLY CHARACTERIZATION WITH SOLID STATE TRACK RECORDERS. Gold, R.; Ruddy, F.H.; Lippincott, E.P.; McElroy, W.N.; Roberts, J.H. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1979. Contract EY-76-C-14-2170. 35p. Dep. NTIS, PC A03/MF A01.

A scoping experiment to characterize the neutron field generated from a Light Water Reactor spent fuel assembly has been successfully completed. Solid State Track Recorder (SSTR) neutron dosimeters were exposed at the surface of a spent fuel assembly from a Pressurized Water Reactor. Acceptable track

- 1825 (HEDL-TME--78-82) NUCLEATION AND

densities were obtained. From these SSTR neutron dosimetry measurements, an absolute neutron flux of about 8×10^3 n/cm² sec was determined at the surface of the spent fuel assembly deduced neutron energy flux, with a mean neutron energy of about 1.3 MeV, is intimately dependent upon the actinide content of the spent fuel.

1828 (HEDL-TME--78-90) SODIUM TECHNOLOGY PROGRESS REPORT, JANUARY--MARCH 1978. Atwood, J.M. (comp.). (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1979. Contract EY-76-C-14-2170. 65p. AT.
This report presents a quarterly summary of progress made at Westinghouse Hanford Company in the areas of LMFBR-related radioactivity control technology and sodium systems technology.

1829 (HEDL-TME--78-91) SODIUM TECHNOLOGY. PROGRESS REPORT, APRIL--JUNE 1978. Atwood, J.M. (comp.). (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1979. Contract EY-76-C-14-2170. 81p. AT.
The report presents a quarterly summary of progress made at Westinghouse Hanford Company in the areas of radioactivity control technology and sodium systems technology.

1830 (HEDL-TME--78-92) SODIUM TECHNOLOGY PROGRESS REPORT, JULY--SEPTEMBER 1978. Atwood, J.M. (comp.). (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1979. Contract EY-76-C-14-2170. 57p. AT.
The report presents a quarterly summary of progress made at Westinghouse Hanford Company in the areas of radioactivity control technology and sodium systems technology.

1831 (HEDL-TME--78-93) SODIUM TECHNOLOGY PROGRESS REPORT, OCTOBER--DECEMBER 1978. Atwood, J.M. (comp.). (Hanford Engineering Development Lab., Richland, WA (USA)). Mar 1979. Contract EY-76-C-14-2170. 67p. AT.
This report presents a quarterly summary of progress made at Westinghouse Hanford Company in the areas of radioactivity control technology and sodium systems technology.

1832 (HEDL-TME--78-99) SPECIMEN SIZE CONSIDERATIONS IN FATIGUE--CRACK GROWTH RATE TESTING. James, L.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Dec 1978. Contract EY-76-C-14-2170. 34p. Dep. NTIS, PC A03/MF A01.

The effect of specimen size upon the fatigue-crack growth behavior of annealed AISI Type 304 stainless steel was studied at two stress ratios at an elevated temperature. The resulting data were examined in the light of the present ASTM size criterion (based upon monotonic yield strength) as well as a proposed criterion based upon the material flow strength. The flow stress criterion is considerably less restrictive than the present ASTM Criterion, and the results of this study indicate that the flow stress criterion is entirely appropriate for strain-hardening materials such as the austenitic stainless steels.

1833 (HEDL-TME--78-101) ELEVATED TEMPERATURE J/SUB IC/ FRACTURE TOUGHNESS BEHAVIOR OF TYPE 304 STAINLESS STEEL AND TYPE 308 STAINLESS STEEL WELDMENT. INTERIM REPORT. Mills, M.J. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1979.

Contract EY-76-C-14-2170. 26p. AT.

The elastic-plastic J/sub IC/ fracture toughness response of FFTF reactor vessel Type 304 stainless steel base metal and Type 308 stainless steel weldment was evaluated at 800°F (427°C) by the multi-specimen R-curve technique. The Type 304 base metal exhibited a high value of J/sub IC/, 3095 in.-lb/in.² (542 kJ/m²), indicating that this austenitic stainless steel possesses very high fracture resistance under elevated temperature conditions. The J/sub IC/ response of the reactor vessel weldment at 800°F (427°C) was slightly less than one-half that exhibited by the base metal. Although the ductile fracture toughness of the weldment was considerably lower than that of the base metal, its fracture resistance was still relatively high compared to those of most structural alloys. Both materials are extremely resistant to unstable fracture even at J levels well above J/sub IC/. This study indicates that a 1-inch (2.5-cm) thick specimen will meet the proposed ASTM size criterion for establishing a valid J/sub IC/ value for the Type 308 weldment at 800°F (427°C), but that a 1.5- to 2-inch (3.8- to 5.1-cm) thick specimen of Type 304 base metal would be required to satisfy the proposed size criterion.

1834 (HEDL-TME--78-104) SODIUM EROSION OF BORON CARBIDE FROM BREACHED ABSORBER PINS. Baker, D.E.; Basmajian, J.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1979. Contract EY-76-C-14-2170. 50p. AT.

The erosion of irradiated boron carbide by flowing sodium was investigated in an out-of-reactor loop test and in an irradiation test conducted in EBR-II. Sodium flow velocity over the intentionally breached absorber pins was 5 ft/sec. Important parameters investigated in the irradiation test were boron carbide burnup, temperature and defect type (hole or slit). Similar erosion behavior was observed for the irradiation test and the loop test. Erosion was not enhanced by the reactor radiation environment. Erosion was confined to the defect area and was influenced by boron-10 burnup in the boron carbide pellets and by temperature.

1835 (HEDL-TME--78-107) CALCULATION OF NEUTRON SOURCE STRENGTH IN FFTF FUEL AS A FUNCTION OF IRRADIATION. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 62p. AT.

Isotope buildup and depletion in reactor fuel as a function of irradiation were evaluated with the ORIGEN and ALCHEMY codes using updated libraries of effective cross sections and half-lives. More accurate evaluations of isotopic density changes in fuel than previously possible were made at the Hanford Engineering Development Laboratory using the Evaluated Nuclear Data File/B, Version V (ENDF/B-V) cross sections. Libraries of one-energy-group effective cross sections for capture, fission and (n,2n) reactions were developed by spectrum-averaging 12 energy group multigroup cross sections with typical 12-energy-group spectra in the inner and outer driver regions of FFTF Cores 1 and 2 at the beginning of life (BOL), the beginning of cycle 4 (BOC-4), and end of cycle 4 (EOC-4). The calculational results of isotope depletion and buildup for inner driver and outer driver fuel were used with recently evaluated neutron yields for spontaneous fission and (α,n) reactions. These provide the most accurate neutron source level evaluation currently obtainable with existing nuclear data.

- 1836 (HEDL-TME--78-108) MELT-IIIB: AN UPDATED VERSION OF THE MELT CODE. Tabb, K.K.; Lewis, C.H.; O'Dell, L.D.; Padilla, A. Jr.; Smith, D.E.; Wilburn, N.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1979. Contract EY-76-C-14-2170. 104p. Dep. NTIS, PC A06/MF A01.

The MELT series is a reactor modeling code designed to investigate a wide variety of hypothetical accident conditions, particularly the transient overpower sequence. MELT-IIIB is the latest in the series.

- 1837 (HEDL-TME--78-111) DECAY HEAT DUE TO ACTIVATION OF CORE COMPONENTS. Luksic, A.T. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1979. Contract EY-76-C-14-2170. 58p. AT.

The activation of the Fast Flux Test Facility reactor core components and the decay heat due to activation products were calculated to provide a basis for handling and storing these materials. The inventory and distribution of radionuclides produced by activation of core components are presented. Activating neutron fluxes were taken from earlier three-dimensional diffusion theory calculations, whereas activation cross-sections are predominantly from the FTR-300S set.

- 1838 (HEDL-TME--78-115) BREEDER REACTOR REFERENCE CONTROL MATERIALS. SEMIANNUAL STATUS NEWSLETTER, JUNE--DECEMBER 1978. Roake, W.E. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1979. Contract EY-76-C-14-2170. 49p. AT.

Progress in the development of LMFBR reference core materials is reported under the following heading: improved absorber concepts, design and performance analysis, materials testing and characterization, feature testing, neutron absorber performance monitoring, and fabrication and procurement.

- 1839 (HEDL-TME--79-1) REAL-TIME ANALYSIS OF FFTF FUEL PIN INSPECTION DATA AT THE ²⁵²Cf ASSAY STATION. Gottschalk, G.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1979. Contract EY-76-C-14-2170. 222p. AT.

The ²⁵²Cf Assay System used at the Hanford Engineering Development Laboratory (HEDL) has been upgraded to provide computerized data acquisition and analysis of total fissile content measurement and fuel column uniformity data. A laser measurement system has been incorporated for automated acquisition of fuel pin diameter data.

- 1840 (HEDL-TME--79-2) ANALYTICAL VALIDATION OF THE CACECO CONTAINMENT ANALYSIS CODE. Peak, R.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1979. Contract EY-76-C-14-2170. 193p. Dep. NTIS PC E05/MF E05.

The CACECO containment analysis code was developed to predict the thermodynamic responses of LMFBR containment facilities to a variety of accidents. This report covers the verification of the CACECO code by problems that can be solved by hand calculations or by reference to textbook and literature examples. The verification concentrates on the accuracy of the material and energy balances maintained by the code and on the independence of the four cells analyzed by the code so that the user can be assured that the code analyses are numerically correct and independent of the organization of the input data submitted to the

code.

- 1841 (HEDL-TME--79-4) IRRADIATION EFFECTS IN FFTF RADIAL REFLECTOR MATERIALS. Ward, A.L.; Blackburn, L.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1979. Contract EY-76-C-14-2170. 44p. AT.

Postirradiation tension tests and swelling measurements were performed on actual FFTF radial reflector materials (316 SS and Alloy 600) irradiated in EBR-II to a total fluence of 2.7×10^{22} n/cm² at temperatures of 514, 566, and 691°C. Ductility results for 316 SS indicate that the portion of reflector lifetime associated with reaching a uniform elongation level of 10% was estimated conservatively by a factor of 2 to 3 in the design. Swelling results at this relatively low fluence were generally consistent with correlations used for reflector design. Data from higher exposure irradiation tests are needed for final verification of reflector design lifetime predictions.

- 1842 (HEDL-TME--79-9) TEST DATA AND POST-TEST ANALYSES OF THREE PROTOTYPIC-LENGTH FFTF FUEL PINS TRANSIENT TESTED IN FLOWING SODIUM (HUT-L2). Bard, F.E.; Gneiting, B.C.; Horn, W.P.; Houser, T.R.; Manthos, E.J.; Randlek, E.H. (Hanford Engineering Development Lab., Richland, WA (USA)). Apr 1979. Contract EY-76-C-14-2170. 180p. AT.

The post-test measurements and the thermal, mechanical, and design procedure analyses of the HUT-L2 transient test are presented. This test was a loop experiment conducted in the TREAT facility. The HEDL MARK-IIC integral sodium loop test vehicle was used to simultaneously test three prototypic-length fresh fuel pins in a simulated FFTF environment. The objectives of the experiment were to obtain measurable cladding deformation and fuel melting for all three fuel pins. Fuel melting occurred in each fuel pin but measurable deformation occurred only in the high smeared density HUT-L2-4 fuel pin.

- 1843 (HEDL-TME--79-17) LIQUID METALS FIRE CONTROL ENGINEERING HANDBOOK. Ballif, J.L. (comp.). (Ford, Bacon and Davis Utah, Inc., Salt Lake City (USA)). Feb 1979. Contract EY-76-C-14-2170. 159p. Dep. NTIS, PC A08/MF A01.

This handbook reviews the basic requirements of the use of liquid metals with emphasis on sodium which has the greatest current usage. It delineates the concepts necessary to design facilities both radioactive and nonradioactive for use with liquid metals. It further reviews the state-of-the-art in fire extinguishers and leak detection equipment and comments on their application and sensitivity. It also provides details on some engineering features of value to the designer of liquid metal facilities.

- 1844 (HEDL-TME--79-19) DUCTILITY AND STRENGTH OF FFTF/CRBRP STRUCTURAL MATERIALS IRRADIATED IN VARIOUS SPECTRA. SECOND INTERIM REPORT. Ward, A.L.; Blackburn, L.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1979. Contract EY-76-C-14-2170. 55p. AT.

Experimental description of the fluence and spectrum dependence of strength and ductility in 304SS, 316SS, weld-deposited 308L SS and Alloy 718 irradiated at low temperatures (644 to 699K) was extended to a total fluence of 3.7×10^{26} n/m². Tension test results for the stainless steel materials from the FFTF/CRBRP Structural Materials Irradiation Experiment

continued to verify earlier indications of significant spectral effects. Application of simple scaling procedures yielded relative damage cross sections which were found to incorporate separable spectrum and temperature dependencies. When normalized to a single temperature (658K) and multiplied by the total fluence, the cross sections provide an excellent correlation parameter for the irradiation damage. Energy-dependent fluence limits for 10% total elongation in the stainless steel materials were determined using the spectral and temperature dependencies of this study coupled with data variability evaluated in a larger data base. Irradiation effects in Alloy 718 were small relative to those observed in the stainless steels and spectral effects could not be defined quantitatively in detail. Estimates of fluence limits for 10% total elongation were determined for moderately soft spectra.

- 1845 (HEDL-TME--79-22) USERS' GUIDE TO CACECO CONTAINMENT ANALYSIS CODE. Peak, R.D. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1979. Contract EY-76-C-14-2170. 216p. Dep. NTIS, PC A10/MF A01.

The CACECO containment analysis code was developed to predict the thermodynamic responses of LMFBR containment facilities to a variety of accidents. The code is included in the National Energy Software Center Library at Argonne National Laboratory as Program No. 762. This users' guide describes the CACECO code and its data input requirements. The code description covers the many mathematical models used and the approximations used in their solution. The descriptions are detailed to the extent that the user can modify the code to suit his unique needs, and, indeed, the reader is urged to consider code modification acceptable.

- 1846 (HEDL-TME--79-23) RESULTS FROM THE RUN-BEYOND-CLADDING-BREACH IRRADIATION OF A PREDEFECTED FUEL PIN (RBCB-6). Langstaff, D.C.; Almasy, M.Y.; Washburn, D.F. (Hanford Engineering Development Lab., Richland, WA (USA)). Aug 1979. Contract EY-76-C-14-2170. 61p. Dep. NTIS, PC A04/MF A01.

A slit was machined through the cladding of an irradiated fuel pin and irradiation in the Experimental Breeder Reactor-II (EBR-II) was resumed. The condition of the fuel pin was continuously followed with delayed neutron (DN) monitors. When the DN signal increased to a previously established administrative limit, the test was terminated. Postirradiation examination showed the sodium-fuel reaction caused fuel swelling and extension of the machined slit. There was no evidence of fuel loss nor was there any indication of impending pin-to-pin failure propagation. This test supports an increase in DN signal for subsequent run-beyond-cladding-breach (RBCB) tests.

- 1847 (HEDL-TME--79-24) SODIUM TECHNOLOGY PROGRESS REPORT, JANUARY--MARCH 1979. Atwood, J.M. (comp.). (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1979. Contract EY-76-C-14-2170. 54p. AT.

This report presents a quarterly summary of progress made at Westinghouse Hanford Company in the areas of radioactivity control technology and sodium systems technology.

- 1848 (HEDL-TME--79-25) SODIUM TECHNOLOGY. PROGRESS REPORT, APRIL-JUNE 1979. Atwood,

J.M. (comp.). (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1979. Contract EY-76-C-14-2170. 55p. AT.

This report presents a quarterly summary of progress made at Westinghouse Hanford Company in the areas of radioactivity control technology and sodium systems technology.

- 1849 (HEDL-TME--79-28) AEROSOL BEHAVIOR DURING SODIUM POOL FIRES IN A LARGE VESSEL: CSTF TESTS AB1 AND AB2. Hilliard, R.K.; McCormack, J.D.; Postma, A.K. (Hanford Engineering Development Lab., Richland, WA (USA)). Jun 1979. Contract EY-76-C-14-2170. 160p. Dep. NTIS, PC A08/MF A01.

Two large-scale aerosol behavior tests, using sodium pool fires as the aerosol source, were performed in the Containment Systems Test Facility (CSTF). The tests were conducted to characterize the properties and behavior of sodium aerosol particles formed and aged in a large containment vessel. The 20-m high, 850-m³ CSTF containment building in regard to parameters that affect agglomeration and gravitational settling. In both tests, sodium burned for one hour in a 4.38-m² pool, and the only difference between them was that steam was injected during the second test, simulating the release of water vapor from heated concrete.

- 1850 (HEDL-TME--79-30) PIPELINES SUPPORTED BY NONLINEAR LOAD-RESISTING MECHANICAL SNUBBERS. Barta, D.A. (Hanford Engineering Development Lab., Richland, WA (USA)). Jul 1979. Contract EY-76-C-14-2170. 48p. AT.

An analytical study of effects of nonlinearities in piping supports on response to seismic excitation is presented. Response calculations for simplified single degree of freedom nonlinear models are used to illustrate sensitivity to stiffness variations, lost motion, and impact damping. Seismic responses of typical spans of various sizes of piping supported by both linear and nonlinear constraints are compared to assess the support load magnifications due to impacting. The idealized nonlinear piping support models are integrated with a finite element model of a large piping system. Time domain seismic responses of the nonlinear piping system are compared to loads determined by a standard linearized seismic response spectra technique.

- 1851 (HEDL-TME--79-34) SIFAIL: A SUBPROGRAM TO CALCULATE CLADDING DEFORMATION AND DAMAGE FOR FAST REACTOR FUEL PINS. Wilson, D.R.; Dutt, D.S. (Hanford Engineering Development Lab., Richland, WA (USA)). May 1979. Contract EY-76-C-14-2170. 137p. Dep. NTIS, PC A07/MF A01.

SIFAIL is a series of subroutines used in conjunction with the thermal performance models of SIEX to assist in the evaluation of mechanical performance of mixed uranium plutonium oxide fuel pins. Cladding deformations due to swelling and creep are calculated. These have been compared to post-irradiation data from fuel pin tests in EBR-II. Several fuel pin cladding failure criteria (cumulative damage, total strain, and thermal creep strain) are evaluated to provide the fuel pin designer with a basis to select design parameters. SIFAIL allows the user many property options for cladding material. Code input is limited to geometric and environmental parameters, with a consistent set of material properties provided by the code. The simplified, yet adequate, thin wall stress-strain calculations provide a reliable estimate of fuel pin mechanical performance, while requiring a small amount of core storage and

computer running time.

- 1852 (HEDL-TME--79-39) INTERIM REPORT FOR THE PINEX-2 TRANSIENT TEST. Porten, D.R.; Padilla, A. Jr.; Randlev, E.H. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1979. Contract EY-76-C-14-2170. 174p. AT.

PINEX-2, the first international benchmark test performed as part of the TREAT/CABRI Exchange, was a dual-objective experiment to continue development of the Pinhole Imaging (PINEX) system for monitoring fuel motion, and to demonstrate the feasibility of internal fuel motion as a mechanism for mitigating the consequences of hypothetical LMFBR accidents. A special fuel pin with a pre-fabricated central hole in the fuel, insulator, and reflector was pre-irradiated in the General Electric Test Reactor and subjected to a \$5/sec transient overpower excursion in the Transient Reactor Test (TREAT) Facility. Approximately 10 grams of molten fuel were ejected upward from the fuel region into the reflector and fission gas plenum.

- 1853 (HEDL-TME--79-42) UNCERTAINTY ANALYSIS OF CONSEQUENCES CALCULATED BY LARGE COMPUTER CODES. Nguyen, D.H. (Hanford Engineering Development Lab., Richland, WA (USA)). Sep 1979. Contract EY-76-C-14-2170. 59p. AT.

A methodology has been constructed to assess the uncertainty in an output consequence calculated by a large code, due to the uncertainties in input data. An application was made to the computer code MELT-IIIA to determine the confidence level in predicting the time of initial pin failure during a TOP accident in the FTR.

- 1854 (HEDL-TME--79-62) RESULTS FROM THE RUN-BEYOND-CLADDING BREACH IRRADIATION OF A PREDEFECTED FUEL PIN (RBCB-7). Langstaff, D.C.; Almasy, M.Y.; Washburn, D.F. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1980. Contract EY-76-C-14-2170. 82p. Dep. NTIS, PC A05/MF A01.

A slit was machined through the cladding of an irradiated fuel pin and irradiation in the Experimental Breeder Reactor-II (EBR-II) was resumed. The condition of the fuel pin was continuously followed with delayed neutron (DN) monitors. When the DN signal increased to a previously established administrative limit of 800 counts per second, the test was terminated. Postirradiation examination showed the sodium-fuel reaction caused fuel pin swelling and extension of the machined slit. There was no evidence of fuel washout nor was there any indication of impending pin-to-pin failure propagation. This test supports an increase in DN signal for subsequent run-beyond-cladding-breach (RBCB) tests.

- 1855 (HEDL-TME--79-72) DILATE: A 2-D STRUCTURAL PROGRAM FOR THE DILATION RESPONSE OF HEXAGONAL DUCTS. Chan, D.P. (Hanford Engineering Development Lab., Richland, WA (USA)). Feb 1980. Contract EY-76-C-14-2170. 48p. Dep. NTIS, PC A03/MF A01.

An analytical method is introduced for determining the dilation of hexagonal ducts in fast reactors. The method, which is valid for temperatures where creep is linearly dependent on stress, was implemented in a fast-running computer called DILATE. A bench-mark program is presented, which shows the results of the DILATE program in close agreement with the results of the finite element program MARC-CDC. User instructions for the DILATE program are

described in detail and a listing of the program is included.

- 1856 (HEDL-TME--1480(FP)) DOSIMETER BASED ADJUSTMENTS OF FAST NEUTRON SPECTRA. Schmittle, F.; Rawlins, J.A. (Hanford Engineering Development Lab., Richland, WA (USA)). 1978. Contract EY-76-C-14-2170. 20p. (CONF-780622--47). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

The task of using integral measurements to adjust or evaluate differential data is a long-standing problem in undetermined problems, as in the adjustment of multigroup neutron spectra based on dosimeter measurements. A general method for solving this problem is described, and its use in realistic problems is illustrated. Many approaches have been used to deal with a variety of well-known difficulties in this area. In particular, modern methods of generalized linear least-squares that include complete data covariance matrices have proved especially useful. The method described is based on an extension of these ideas. 6 figures, 1 table. (RWR)

- 1857 (HWS--1880(Rev.2)(Add.1)) FORCED CIRCULATION COLD TRAP ASSEMBLIES FOR REMOVAL OF SODIUM IMPURITIES FROM FFTF CLOSED LOOP SYSTEMS. Kios, D.B. (Hanford Engineering Development Lab., Richland, WA (USA)). 1975. Contract EY-76-C-14-2170. 124p. AT.

Revised specifications are presented for the FFTF closed loop forced circulation cold trap assemblies.

- 1858 (HWS--2160) FIRST ARTICLE TEST REPORT FOR FFTF HEATER SUPPORTS. (Westinghouse Electric Corp., Pittsburgh, PA (USA)). Astronuclear Lab.). 23 Dec 1974. Contract EY-76-C-14-2170. 137p. Dep. NTIS, PC A07/MF A01.

Test articles of 1-in. and 28-in. diameter have been assembled, tested in accordance with HWS-2160 and the FAT Plan, Revision 1, and disassembled and evaluated in the presence of a Bechtel representative. Observations and conclusions are presented.

- 1859 (IAE--2842) INTERFERENCE OF NEUTRON-ABSORBING CONTROL RODS IN A LARGE POWER FAST REACTOR. Karpov, V.A.; Matveev, V.I.; Protzenko, A.N. (Gosudarstvennyy Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Moscow, Inst. Atomnoj Ehnergii). 1977. 13p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

A numerical experiment investigating interference of absorbing rods in a BN-600-type reactor using the BFS-24-16 critical assembly is considered. The experiment was carried out with two-dimensional KENT program based on the grid solution technique of the two-group diffusion equation in hexagonal geometry. It is shown that in case of the optimum location of rods the gain up to 20% can be obtained in the system effectiveness. Using an example of internal and external rings of the absorbing rods it is shown that the methodical error of pairing interference is equal to 10% of the interference effect.

- 1860 (IAEA--207, pp 353-357) SOME REQUIREMENTS AS REGARDS THE ACCURACY OF CONSTANTS FOR FAST POWER REACTOR SHIELDING CALCULATIONS. Bakumenko, O.D.; Kulakovsky, M.Ya.; Savitsky, V.I. 1978.

From Specialists meeting on nuclear data requirements for shielding; Vienna, Austria (8 Oct 1976).

The intrinsic activity of sodium coolant, determined by the sodium activation components resulting from the reactions $^{23}\text{Na}(n,\gamma)^{24}\text{Na}$ and $^{23}\text{Na}(n,2n)^{22}\text{Na}$, was determined when the BN-350 reactor started up. Another series of experiments in the BN-350 reactor was carried out for the purpose of determining the activity of the steel components resulting from reactions $^{50}\text{Cr}(n,\gamma)^{51}\text{Cr}$, $^{58}\text{Fe}(n,\gamma)^{59}\text{Fe}$, $^{58}\text{Ni}(n,p)^{58}\text{Co}$, $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ and $^{59}\text{Co}(n,\gamma)^{60}\text{Co}$. Comparison of calculated and experimental values are made. Some discrepancies are pointed out.

1861 (IAEA--207, pp 245-259) SENSIBILITY STUDIES OF THE EQUIVALENT THERMAL NEUTRON FLUX ON THE HEAT EXCHANGER OF A SODIUM COOLED FAST REACTOR. (1. PT.). Boioli, A.; Fiorini, L. (NIRA); Moreau, J.; Suntinger, G. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France)). 1978. (In French).

From Specialists meeting on nuclear data requirements for shielding; Vienna, Austria (8 Oct 1976).

This paper reports on sensibility studies of the equivalent thermal neutron flux on the heat exchanger for a sodium cooled fast reactor. Graphs and diagrams of the neutron flux in function of the reactor geometry, contribution of the fission sources in the core and the blanket of the reactor are given.

1862 (IAEA--207, pp 431-463) RESULTS OF NEUTRON-PROPAGATION EXPERIMENTS PERFORMED IN IRON-SODIUM MIXTURES USING THE SOURCE REACTOR HARMONIE AND TAPIRO. Bouteau, F.; Calamand, D.; Ocerasies, Y.; Vienot, R. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France)); Antonini, D.; Bozzi, L.; Martini, M.; Moio, P. (Comitato Nazionale per l'Energia Nucleare, Casaccia (Italy). Centro di Studi Nucleari). 1978. (In French).

From Specialists meeting on nuclear data requirements for shielding; Vienna, Austria (8 Oct 1976).

The results of neutron-propagation experiments performed in iron-sodium mixtures, using the source reactor HARMONIE and TAPIRO are presented: with HARMONIE: five media were studied: pure sodium and pure iron, and mixtures of iron and sodium (30 V/o, 50 V/o, 70 V/o); with TAPIRO the presently available results concern a pure sodium medium; two types of results are given: first, the raw results; then, the correction factors for heterogeneity and leakage to convert the raw results into the ideal homogeneous and spherical model used in the calculations. Finally, a comparison is made between results obtained with HARMONIE and TAPIRO for the pure sodium; the aim of this comparison is to verify that the differences in geometry and source spectra are properly accounted for the calculations.

1863 (IAEA--207, pp 465-504) ANALYSIS OF NEUTRON PROPAGATION EXPERIMENTS IN IRON-SODIUM MIXTURES WITH THE PROPANE CALCULATIONAL SCHEME FOR FAST REACTORS. Estiot, J.C.; Trapp, J.P. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France)); Salvatore, M.; Palmiotti, G. (Comitato Nazionale per l'Energia Nucleare, Casaccia (Italy). Centro di Studi Nucleari). 1978. (In French).

From Specialists meeting on nuclear data requirements for shielding; Vienna, Austria (8

Oct 1976).

PROPANE is a calculational scheme for neutron propagation in the shield of fast reactors. PROPANE 0 is the first, non adjusted version of this scheme. Results of integral neutron propagation experiments obtained in pure iron, pure sodium, and iron-sodium mixtures (HARMONIE-TAPIRO and ASPEX experiments) are interpreted using PROPANE 0. Experimental results are, when necessary, corrected for heterogeneity and leakage and converted into a simple homogeneous one dimensional model before comparison with calculations. For HARMONIE and TAPIRO, the calculations consist of propagating a given surface source at the entry of the assemblies. The spectrum of this source is adjusted using proton-recoil and spectral indices measurements. The present version of PROPANE 0 is assessed. Possible source of discrepancies are discussed using essentially the results of sensitivity studies.

1864 (IAEA--207, pp 505-530) TWO-DIMENSIONAL SHIELDING BENCHMARKS FOR IRON AT YAYOI (I). Oka, Y.; An, S.; Kasai, S.; Yoshii, R.; Hashikura, H.; Akiyama, M. (Tokyo Univ. (Japan)); Miyasaka, S.; Koyama, K. (Japan Atomic Energy Research Inst., Tokai, Ibaraki. Tokai Research Establishment); Hyodo, T. (Kyoto Univ. (Japan)). 1978.

From Specialists meeting on nuclear data requirements for shielding; Vienna, Austria (8 Oct 1976).

The aim of this work is to assess the collapsed neutron and gamma multigroup cross sections for two dimensional discrete ordinate transport code. Two dimensional distributions of neutron flux and gamma ray dose through a 70cm thick and 94cm square iron shield were measured at the fast neutron source reactor "YAYOI". The iron shield was placed over the lead reflector in the vertical experimental column surrounded by heavy concrete wall. The detectors used in this experiment were threshold detectors In, Ni, Al, Mg, Fe and Zn, sandwich resonance detectors Au, W and Co, activation foils Au for neutrons and thermoluminescence detectors for gamma ray dose. The experimental results were compared with the calculated ones by the discrete ordinate transport code ANISN and TWOTRAN. The region-wise, coupled neutron-gamma multigroup cross sections (100n+20gamma, EURLIB structure) were generated from ENDF/B-IV library for neutrons and POPOP4 library for gamma ray production cross sections by using the code system RADHEAT. The effective microscopic neutron cross sections were obtained from the infinite dilution values applying ABBN type self-shielding factors. The gamma ray production multigroup cross sections were calculated from these effective microscopic neutron cross sections. For two-dimensional calculations the group constants were collapsed into 10 neutron groups and 3 gamma groups by using ANISN.

1865 (IAEA--207, pp 221-241) BENCHMARK PROBLEMS FOR THE SHIELDINGS OF A FAST NEUTRON REACTORS. Estiot, J.C.; Trapp, J.P. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France)). 1978. (In French).

From Specialists meeting on nuclear data requirements for shielding; Vienna, Austria (8 Oct 1976).

The BENCHMARK calculations are performed using the calculational scheme (formulaire) for fast reactor shielding PROPANE 0, i.e.: 26 groups, P₁, S₁; the spatial meshes recommended in the BENCHMARK specifications: non adjusted

cross sections, derived from the 100g DLC2 cross sections set (ENDF/BII); the source is the surface anisotropic source given in the BENCHMARK specifications condensed to 26 groups; the calculated integral responses are those of the BENCHMARK specifications, i.e.: Total flux, equivalent thermal flux, flux above 800 and 111 Kev, damage rate in steel, capture rate of sodium, Sulphur and Rhodium responses, Manganese and Sodium responses, U235 fission rate and ^{59}Co capture rate.

- 1866 (IAEA--208(Vol.1), pp 147-181)
SPECTRAL CHARACTERIZATION BY COMBINING NEUTRON SPECTROSCOPY, ANALYTICAL CALCULATIONS, AND INTEGRAL MEASUREMENTS. McElroy, W.N.; Gold, R.; Lippincott, E.P. (Hanford Engineering Development Lab., Richland, Wash. (USA)); Fabry, A. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)); Roberts, J.H. (Macalester Coll., Saint Paul, Minn. (USA)). 1978.

From Consultants meeting on integral cross section measurements in standard neutron fields; Vienna, Austria (15 Nov 1976).

Status and trends of differential spectrometry, calculations, and integral measurements for characterizing neutron spectra are assessed. Emphasis is placed upon the state-of-the-art accuracy that has actually been attained in standard, reference, and controlled-environment benchmark neutron fields. The utility of SAND-II unfolding is stressed for the comparison of differential spectrometry, integral measurements and calculations. Selective adjustments of both spectra and cross sections are still necessary, even for the most advanced benchmark fields and best known cross sections.

- 1867 (IAEA--208(Vol.2), pp 191-208)
INTERCOMPARISON OF THE INTERMEDIATE-ENERGY STANDARD NEUTRON FIELDS AT THE NISUS AND MOL-ΣΣ FACILITIES BY MEANS OF ABSOLUTE FISSION CHAMBERS. Fabry, A. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)); Williams, J.G.; Hannan, A.H.M.A. (London Univ. (UK)). Sunninghill Reactor Centre; Azimi-Garakanl, D. (Teheran Univ. (Iran)). Inst. of Nuclear Science and Technology). 1978.

From Consultants meeting on integral cross section measurements in standard neutron fields; Vienna, Austria (15 Nov 1976).

Intermediate-energy standard neutron fields simulating core neutron spectra of fast breeder reactors are in operation at the Mol-ΣΣ (Belgium), ITN-ΣΣ (Rumania) and NISUS (Great Britain) facilities. These three assemblies are implemented within 50cm diameter spherical cavities hollowed out from conventional graphite thermal columns; the standard configuration consists of an approximately 5cm thick natural uranium spherical source, approximately 25cm outer diameter, surrounding a concentric natural boron carbide shell at the centre of which the standard neutron field is generated. Measurements of absolute fission rates and ratios have been performed in NISUS by means of NBS-type, double gas-flow ionization fission chambers previously exposed in Mol-ΣΣ and ITN-ΣΣ. Leakage of subcadmium neutrons through the access hole into the NISUS standard field has been detected and eliminated. Expected, small neutronic differences between the ΣΣ and NISUS assemblies have been assessed by transport theory computations for comparison with experimental data. The unperturbed fission cross section ratios measured at NISUS centre are (1.000: 1.175 ± 0.027: 0.0568 ± 0.0015: 0.383 ± 0.011) for ^{235}U : ^{239}Pu : ^{238}U : ^{237}Np respectively; they agree with similar data

obtained at the centre of the Mol-ΣΣ and ITN-ΣΣ facilities.

- 1868 (IAEA-AG--63-1) TECHNIQUES FOR SHEARING AND DISSOLVING LMFBR FUELS. Burch, W.D.; Watson, C.D.; Unger, W.E. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 35p. (CONF-760576--32). Dep. NTIS (US Sales Only), PC A03/MF A01.

From IAEA meeting on LMFBR fuel reprocessing; Leningrad, USSR (17 May 1976).

The present paper describes development efforts and the present status of the head-end steps of fuel-element cleaning, shearing, sodium deactivation, and dissolution.

- 1869 (IAEA-AG--63-3) REVIEW OF UNITED STATES PROGRAM IN LMFBR FUEL REPROCESSING. McVey, W.H. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 3p. (CONF-760576--25). Dep. NTIS (US Sales Only), PC A02/MF A01.

From IAEA meeting on LMFBR fuel reprocessing; Leningrad, USSR (17 May 1976).

The overall objective of the LMFBR program is to develop a broad technological and engineering base for a competitive commercial breeder industry as a means for meeting national energy needs in the 1990s and beyond. The review briefly considers the different aspects (industrial capability, public acceptance etc.) that have to be taken into consideration in connection with this. The work required to accomplish the LMFBR program has been divided into seven technology areas (components, materials, physics, chemistry, safety, fuels and fuel recycle) comprising the LMFBR Base Program, and a closely related Project part covering design, construction and operation of major reactor facilities for the LMFBR program. The LMFBR fuel reprocessing developmental plan, as a support effort in the Base Program, is considered more in detail. The objective of this plan is to develop and demonstrate reprocessing systems that accommodate all candidate fuels and allow for the rapid fuel recovery and turnaround times necessary to ensure doubling times of 10 to 15 years.

- 1870 (IAEA-AG--63-4) FAST REACTOR FUEL REPROCESSING IN THE UNITED KINGDOM. Pugh, O. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 7p. (CONF-760576--28). Dep. NTIS (US Sales Only), PC A02/MF A01.

From IAEA meeting on LMFBR fuel reprocessing; Leningrad, USSR (17 May 1976).

The reprocessing activities at Dounreay in the United Kingdom is presented. In 1958 a plant was built to permit recycling of unburnt ^{235}U from the U/Mo metallic fuel element used in the Dounreay Fast Reactor (DFR). The decision to reprocess the irradiated Pu/U oxide fuel from PFR at Dounreay to provide data for the construction of larger reprocessing plants required for commercial fast reactor exploitation, has required considerable modification of the present DFR plant. The modifications include fuel handling, containment and waste management. Special attention is also paid to problems connected with solvent extraction. A brief description of the practical measures taken to carry out the modifications is given.

- 1871 (IAEA-AG--63-5) REPROCESSING OF 'FAST' FUEL IN FRANCE. Sauteron, J.; Boudry, J.C.; Bourgeois, M.; Le Bouhellec, J.;

Miquel, P. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 18p. (In French). Dep. NTIS (US Sales Only), PC A03/MF A01.

The results of laboratory studies as well as pilot testing (AT-I La Hague, Marcoule, Fontenay-aux-Roses) in reprocessing of fast breeder reactor fuels are described. The paper covers all steps: head end, aqueous and fluoride volatility processes, and waste treatment. In conclusion, it is demonstrated why it is still too early to define a strategy of industrial reprocessing for this reactor type.

- 1872 (IAEA-AG--63-6) DEVELOPMENT AND STATUS OF LMFBR FUEL REPROCESSING IN THE FEDERAL REPUBLIC OF GERMANY. Baumgaertner, F.; Ochsenfeld, W. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 28p. (CONF-760576--19). Dep. NTIS (US Sales Only), PC A03/MF A01.

From IAEA meeting on LMFBR fuel reprocessing; Leningrad, USSR (17 May 1976).

The problems associated with reprocessing oxide LMFBR fuel elements result from the higher plutonium content, the higher fission product content and the higher level of thermal decay power; their consequences on the Purex process are well recognizable. Reprocessing LMFBR elements by the conventional Purex process technique is considered to be feasible. If the cooling time of LMFBR fuel elements amount to one year or more, it is only the higher plutonium content which determines the process flowsheet. The necessary chemical steps for reprocessing co-precipitated oxides have been developed at Karlsruhe for this case. The mechanical head end still needs to be developed on a technical scale, as does the management of safety against criticality. If reprocessing is to follow a cooling time of less than 200 days, as envisaged in the original planning for LMFBR fuel elements, the efficiency of offgas treatment and the reliable operation of fast contactors in the first extraction cycle still have to be tested.

- 1873 (IAEA-AG--63-7) CONSIDERATIONS ON THE GAMMA ACTIVITY OF THE PU PRODUCT ARISING FROM LMFBR'S FUEL REPROCESSING. Moccia, A. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 5p. (CONF-760576--26). Dep. NTIS (US Sales Only), PC A02/MF A01.

From IAEA meeting on LMFBR fuel reprocessing; Leningrad, USSR (17 May 1976).

The composition of Pu product and its gamma activity and gamma power were theoretically calculated with reference to the fuel to be recovered from the PEC fast experimental reactor. Preliminary evaluation derived from this calculation allows some conclusions to be drawn related to the extent of the purification from fission products required in the reprocessing step.

- 1874 (IAEA-AG--63-8) HEAD-END AND NUCLEAR GAS PURIFICATION RESEARCH ON LMFBR FUEL REPROCESSING IN BELGIUM. Baetsle, L.H. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 11p. (CONF-760576--18). Dep. NTIS (US Sales Only), PC A02/MF A01.

From IAEA meeting on LMFBR fuel reprocessing; Leningrad, USSR (17 May 1976).

In the framework of the international DEBENELUX project on fast reactor development considerable attention was devoted to the reprocessing problems. The report describes the specific Belgian contributions which were

focused on experimental and conceptual studies of de-activation and transportation, on dissolution of high burn-up MOX fuel, and on technological development of gas purification methods applicable to LMFBR fuel.

- 1875 (IAEA-AG--63-12) PERSPECTIVES OF THE NUCLEAR POWER DEVELOPMENT IN THE USSR AND THE STRATEGY OF FUEL REPROCESSING. Dergachev, N.P. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 10p. (In Russian). (CONF-760576--34). Dep. NTIS (US Sales Only), PC A02/MF A01.

From IAEA meeting on LMFBR fuel reprocessing; Leningrad, USSR (17 May 1976).

The forecast of the USSR program for nuclear power development, the role of fast reactors, the time of their introduction and their effect on fuel cycles are treated in the paper together with the consideration of technical characteristics of different fast reactors and their effect on uranium demand. The basic principles of development of reprocessing capacity is also described.

- 1876 (IAEA-AG--63-13) PHYSICAL CHARACTERISTICS OF SPENT LMFBR FUEL. Bakumenko, O.D.; Ikhlov, E.M.; Kulakovskij, M.Ya.; Troyanov, M.F.; Tsikunov, A.G. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 11p. (In Russian). (CONF-760576--30). Dep. NTIS (US Sales Only), PC A02/MF A01.

From IAEA meeting on LMFBR fuel reprocessing; Leningrad, USSR (17 May 1976).

The present paper describes physical characteristics of fast breeder reactor fuels and analyses calculated isotope composition of spent fuel and dependance of fission product activity on cooling time. Requirements for a decontamination factor from fission product is considered in connection with the re-fabrication of fuel.

- 1877 (IAEA-AG--63-14) SOME ASPECTS OF CLEANING AND TRANSPORT OF SPENT FAST REACTOR FUELS. Shirin, V.M.; Karpov, A.V. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 4p. (In Russian). (CONF-760576--38). Dep. NTIS (US Sales Only), PC A01/MF A01.

From IAEA meeting on LMFBR fuel reprocessing; Leningrad, USSR (17 May 1976).

Cleaning and transport of spent fast reactor fuels and utilization of lead for this purposes are considered in this paper.

- 1878 (IAEA-AG--63-15) FAST REACTOR SPENT FUELS TRANSPORTATION PROBLEMS. Burlakov, V.A.; Kosarev, Yu.A.; Markovin, A.P. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 6p. (In Russian). (CONF-760576--33). Dep. NTIS (US Sales Only), PC A02/MF A01.

From IAEA meeting on LMFBR fuel reprocessing; Leningrad, USSR (17 May 1976).

In this paper transportation problems of spent fuels from fast reactors BN-350, BN-600 and BN-1500 to reprocessing plant are discussed. It was calculated that 6-10 fuel assemblies can be inserted in one transport cask, the weight of which is 50-60t. Noble gas or water can be used for the BN-350 fuel cask; sodium, lead and special alloys for the BN-1500. Transportation problems of angermetized fuel are also considered in this paper.

- 1879 (IAEA-AG--63-16) THERMAL DECLADDING OF BOR-60 FUEL ELEMENTS. Bibikov, S.E.;

Vasil'ev, G.I.; Ivanov, P.M.; Novoselov, G.P.; Serov, A.V.; Skiba, O.V.; Barkov, V.A. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 20p. (In Russian). (CONF-760576--31). Dep. NTIS (US Sales Only), PC A02/MF A01.

From IAEA meeting on LMFBR fuel reprocessing; Leningrad, USSR (17 May 1976).

Experimental results on thermal decladding of reactor BOR-60 fuel elements irradiated to >10% and cooling time - 6 and 3 months - are discussed. The direct yield of fuel is 99.9%; content of steel in fuel <5.4%; data on fission products distribution are also given.

1880 (IAEA-AG--63-17) STORAGE OF FAST REACTOR SPENT FUEL AT THE REACTOR SITE AND IN REPROCESSING PLANTS. Burlakov, V.A.; Kruglikov, G.V.; Saaginskii, B.N. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 5p. (In Russian). (CONF-760576--36). Dep. NTIS (US Sales Only), PC A02/MF A01.

From IAEA meeting on LMFBR fuel reprocessing; Leningrad, USSR (17 May 1976).

The paper considers the following problems: in-vessel spent fuel storage, intermediate storage in sodium, sodium removal and decontamination, utilization of special hermetically sealed containers to avoid release of radioactivity to the storage facility.

1881 (IAEA-AG--63-21) PILOT PLANT REPROCESSING OF THE BOR-60 SPENT URANIUM FUEL BY USING FLUORIDE VOLUTILITY METHOD. Kikoin, I.K.; Dem'yanovich, M.A.; Ivanov, P.M.; Prusakov, V.N.; Serov, A.V.; Skiba, O.V.; Barkov, V.A. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 17p. (In Russian). (CONF-760576--24). Dep. NTIS (US Sales Only), PC A02/MF A01.

From IAEA meeting on LMFBR fuel reprocessing; Leningrad, USSR (17 May 1976).

Reactor BOR-60 spent fuels with burn-up approximately 10% and cooling time 3 and 6 months were successfully processed by the fluoride volatility method, followed up by sorption purification from fission products. The decontamination factor from gamma-radioactivity was >10⁴, fuel losses 0.4-0.5%. The main quantity (85%) of fission products is accumulated in an apparatus for fluorination as a waste; radioactive gas release is less than 10⁻⁵%.

1882 (IAEA-AG--63-27) REPORT OF THE IAEA ADVISORY GROUP MEETING ON LMFBR FUEL REPROCESSING. (International Atomic Energy Agency, Vienna (Austria)). May 1976. 31p. (CONF-760576--23). Dep. NTIS (US Sales Only), PC A03/MF A01.

From IAEA meeting on LMFBR fuel reprocessing; Leningrad, USSR (17 May 1976).

A summary of the papers and discussions of the meeting is presented, reviewing the status of development in LMFBR fuel reprocessing and focusing attention on important problem areas. The following topics are discussed: Transport, storage and removal of sodium; decladding and shearing; dissolution; Purex process; fluoride volatility method; off-gas purification; waste disposal. Status reports of national programmes of Belgium, France, Federal Republic of Germany, Italy, Japan, United Kingdom, USSR and USA are included.

1883 (IAEA-SM--225/24) DESIGN OF THE CLINCH RIVER BREEDER REACTOR STEAM GENERATORS. Lillie, A.F. (Atomics International Div., Canoga Park, CA (USA)). 1978. Contract EY-

76-C-03-1142. 20p. (CONF-780407--2). Dep. NTIS, PC A02/MF A01.

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The design features of the U.S. Clinch River Breeder Reactor Steam Generators are presented. The steam generator system arrangement and the design requirements for the units themselves are discussed. Special design features are discussed in some detail, and a summary of the supporting developments tests which have confirmed the steam generator design is presented.

1884 (IAEA-SM--244/66) CORE DESIGN AND OPTIMIZATION OF HIGH PERFORMANCE LOW SODIUM VOID 1000 MWE HETEROGENEOUS OXIDE LMFBR CORES. Barthold, W.P.; Orzechwa, Y.; Su, S.F.; Beitel, J.C.; Turski, R.; Lam, P.S.K.; Fuller, E.L. (Argonne National Lab., IL (USA); Electric Power Research Inst., Palo Alto, CA (USA)). 1979. Contract W-31-109-ENG-38. 38p. (CONF-790933--5). Dep. NTIS, PC A03/MF A01.

From IAEA international symposium on fast reactor physics; Aix-en-Provence, France (24 Sep 1979).

Radially heterogeneous core configurations are effective means to reduce sodium void reactivity. In general, radially heterogeneous cores can be designed as tightly or loosely coupled cores with center core or center blanket arrangements. Core height, number of core regions and number of fuel pins per assembly are additional variables in an optimization of basic heterogeneous core configurations. An extensive study was carried out to optimize the core configurations for 1000 MWe LMFBRs. All cores were subject to a common set of nuclear, mechanical, and thermal-hydraulic design assumptions. They were restrained by an upper sodium void reactivity limit of \$2.50 and a doubling time of approximately 15 to 18 years. The screening and optimization procedures employed lead to two core layouts which were both tightly coupled. A complete nuclear analysis of these two cores (derived from a loosely coupled configuration/derived from a tightly coupled configuration) determined the fissile inventories (4268.4/4213.4 kg at EOEC), burnups (83.90/100.7 Mwd/t peak), reactivity swings (0.49/1.8% Δk total), power and flux distributions for different control insertion patterns, the breeding performance (15.7/15.3 yrs CSDT), the safety parameters, such as sodium void reactivity (\$2.38/\$2.23 at EOEC), isothermal Doppler coefficients for both sodium-in (45.6/46.1 T dk/dT x 10⁻⁴ core at EOEC) and sodium-out conditions (28.6/28.2 T dk/dT x 10⁻⁴ core at EOEC), and the transient behavior which shows very little space-dependence during a 60 cent reactivity step insertion.

1885 (ICP--1050-4) FAST REACTOR FISSION YIELDS FOR ²⁴⁰Pu AND ²⁴²Pu. Maeck, W.J.; Eggleston, R.L.; Erikson, A.L.; Tromp, R.L. (Allied Chemical Corp., Idaho Falls, ID (USA). Idaho Chemical Programs - Operations Office). Feb 1979. Contract EY-76-C-07-1540. 57p. Dep. NTIS, PC A04/MF A01.

Absolute fast reactor fission yield data are presented for ²⁴⁰Pu and ²⁴²Pu irradiated in EBR-II. The measurement technique was isotope dilution mass spectrometry for over 40 mass numbers.

1886 (ICP--1050-11) FAST REACTOR FISSION YIELDS FOR ²³⁹Pu AND ²⁴¹Pu. Maeck, W.J. (Idaho National Engineering Lab., Idaho Falls

- (USA)). Aug 1977. Contract EY-76-C-07-1540. 58p. Dep. NTIS, PC A03/MF A01.
- Fast fission yield data are presented for ^{239}Pu and ^{241}Pu irradiated in EBR-II. The measurement technique was isotope dilution mass spectrometry. The neutron environment for the irradiation is defined, and chemical separation and mass spectrometric procedures are briefly described. Recommendations and suggestions for the selection of burnup monitors are given.
- 1887 (ICP--1168) SECOND ELECTROLYTIC DISSOLUTION CAMPAIGN OF EBR-II FUEL AT ICPP. Mortimer, S.R. (Allied Chemical Corp., Idaho Falls, ID (USA). Idaho Chemical Programs - Operations Office). Dec 1978. Contract EY-76-C-07-1540. 35p. Dep. NTIS, PC A03/MF A01.
- The second fuel reprocessing campaign using the electrolytic dissolution facility at the Idaho Chemical Processing Plant (ICPP) took place from February 1975 to November 1976. Several forms of EBR-II fuel, and various scrap fuels containing highly-enriched uranium, were dissolved in nitric acid in the dissolver's electric field. Gadolinium was used as a soluble nuclear poison to guarantee nuclear safety in the headend processing equipment. 9 figures, 5 tables.
- 1888 (IDO--1570-T31) INVESTIGATION OF OPTIMAL REDUNDANCY FOR A SPECIFIC SYSTEM OBJECTIVE. Griffith, J.M.; Rasmuson, D.M. (EG and G Idaho, Inc., Idaho Falls (USA)). 1979. Contract EY-76-C-07-1570. 23p. Dep. NTIS, PC A02/MF A01.
- Component redundancy is an effective technique for reducing system unavailability. However, the mathematical models relating redundancy and unavailability are generally complex, and casual inspection does not reveal which components should be made redundant to reduce unavailability. This paper develops a method for determining which components are made redundant with the greatest concomitant unavailability reduction. The method was used to study three system designs. For each design it was possible to identify several components which could be made redundant with an associated reduction in unavailability. However, the method also demonstrated that making most components redundant does not reduce unavailability, and in one case a component redundancy actually increased unavailability. Application of the present method helps the designer develop insight concerning the role of component redundancy in a system design without complex calculations and interpretations. 6 figures, 1 table.
- 1889 (IKE--4-87) BENCHMARK CALCULATIONS FOR A FAST SODIUM-COOLED BREEDER REACTOR ACCORDING TO THE METHOD OF FINITE ELEMENTS. Arnold, C.; Schmidt, F. (Stuttgart Univ. (TH) (Germany, F.R.). Inst. fuer Kernenergetik und Energiesysteme). Feb 1979. 30p. (In German). Dep. NTIS (US Sales Only), PC A03/MF A01.
- For a fast sodium-cooled breeder reactor a benchmark problem is calculated two- and three-dimensional according to the method of finite elements. The results are valued and compared with results obtained according to the method of finite differences. For the two-dimensional case good agreement is obtained. For the upper half of the core the results differ by .02%, for the lower half by .009%. For the three-dimensional case they differ by .07%. Also good agreement is obtained for flux comparison. The results are obtained with a smaller consumption of computing time.
- 1890 (IKE--5-202) COMPUTATION OF TEMPERATURE AND VELOCITY FIELDS IN FUEL ELEMENTS OF FAST SODIUM-COOLED BREEDERS PASSED BY A PARALLEL FLOW. Straub, G. (Stuttgart Univ. (TH) (Germany, F.R.). Inst. fuer Kernenergetik und Energiesysteme). Oct 1976. 93p. (In German). Dep. NTIS (US Sales Only), PC A05/MF A01.
- Thesis.
- A factor of decisive importance for the safe operation of nuclear reactors is the knowledge of the precise temperature distribution in the fuel elements. This study is to improve the computation of velocity and temperature fields in hexagonal fuel elements passed by a sodium flow. For this purpose, the turbulent transport phenomena in the turbulent bundle flow and their repercussions on the velocity and temperature fields had to be investigated. The ARTIS computer program was developed for calculation of the velocity and temperature fields; its results are compared with four different experiments.
- 1891 (IKE--5-203) DESIGN AND OPTIMIZATION OF THE ELECTROMAGNETICALLY ENERGIZED LIQUID METAL DIRECT CURRENT CONDUCTION PUMPS. Steiner, D. (Stuttgart Univ. (TH) (Germany, F.R.). Inst. fuer Kernenergetik und Energiesysteme). Dec 1976. 152p. (In German). Dep. NTIS (US Sales Only), PC A08/MF A01.
- Thesis.
- For direct-current conduction pumps with electromagnetic field-energizing an optimization method is developed accounting for the effects of electrical geometrical and materials specific quantities in the form of dimensionless power products. With fixed pump geometry the conditions for the maximum efficiency, including Joule's losses in the transformer channel, in the field windings, and in the electric power supplies are first set up. In the case of series excitation, minimizing Joule's losses in the power supplies and field windings have the effect of tending towards a design for small currents. With the aid of dimensionless geometry parameters the effects of variable pump dimensions are superimposed upon the electrical design. The optimum design can be represented as an envelope of a multi-parameter set of curves in a mass-efficiency diagram.
- 1892 (INB--414) MONTHLY REPORT NO. 9 (SEPTEMBER, OCTOBER, AND NOVEMBER 1977) OF INTERATOM REPRESENTATIVE TO WESTINGHOUSE HANFORD CO., RICHLAND, WASHINGTON (USA). Mayer, H. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). 30 Nov 1977. Contract EY-76-C-14-2170. 41p. Dep. NTIS (US Sales Only), PC A03/MF A01.
- The following topics are mentioned: alternate breeder fuel cycles (Shippingport Light Water Breeder Reactor), Clinch River Breeder Reactor status, FFTF status, LMFBF component development, and PLBR status. (DG)
- 1893 (INIS-mf--4052, pp 67-69) SIMULATING THE LMFBF NEUTRONICS AND PRIMARY HEAT TRANSPORT WITH DNSP. Saphier, D. (Israel Atomic Energy Commission, Yavne. Soreq Nuclear Research Center). 1977.
- From Meeting of the Israel Nuclear Societies Technion; Haifa, Israel (4 Dec 1977).
- HEAT TRANSFER;LMFBF TYPE REACTORS; MATHEMATICAL MODELS;PROGRAMMING;REACTIVITY; REACTOR KINETICS;SIMULATION

- 1894 (INIS-mf--4054, pp 182-192) RESULTS OF RESEARCH INTO NUCLEAR POWER PLANT SAFETY. Polak, V.; Hladky, E.; Moravek, J.; Suchomel, J.; Stehlik, J. Sep 1976. (In Slovak).

From 25. anniversary of the Power Research Institute; Prague, Czechoslovakia (30 Sep 1976).

A survey is given of computer programmes for the safety analysis of nuclear power plants with WWR type reactors and with fast breeder reactors. The programmes solve accidents in the core, the primary circuit and the containment. A comparison is made of Czechoslovak and foreign computer programmes and their agreement proved. Also studied is the problem of radiation safety of nuclear power plants with regard to the leakage of radioactive isotopes and their detection.

- 1895 (INIS-mf--4280, pp 163-166) PROGRESS IN LIQUID METALS HEAT TRANSFER. Pfann, J. 1976. (In Czech).

From Scientific and technical conference commemorating the 20. anniversary of the Nuclear Research Institute's foundation; Rez, Czechoslovakia (10 Jun 1975).

A mathematical model was drafted for heat transfer in liquid metals in turbulent flow. It was shown that the relation $Nu=Nu(Re, K_1, \dots, K_{sub}(n))$ applies for the flow region with negligible turbulent heat transfer for the practically most important case of heat transfer in channels with a coolant with $Pr \ll 1$ at fully developed velocity and temperature profiles. In the relation, K_1 to $K_{sub}(n)$ represent dimensionless geometric and physical parameters. In the region with an increased effect of turbulent heat transfer the relation $Nu=Nu(Pr, Re, K_1, \dots, K_{sub}(n))$ applies while for the region with a prevalent effect of turbulent heat transfer the relation $Nu=Nu(Pr, K_1, \dots, K_{sub}(n))$ applies. On the basis of the above model generally valid conclusions were formulated for heat transfer by liquid metals in a longitudinally by-passed bundle of fuel rods forming a regular triangular lattice.

- 1896 (INIS-mf--4280, pp 69-73) CALCULATIONS OF PHYSICS CHARACTERISTICS OF THE FAST REACTOR CORE. Cermak, J.; Tinka, I.; Tinkova, E. 1976. (In Czech).

From Scientific and technical conference commemorating the 20. anniversary of the Nuclear Research Institute's foundation; Rez, Czechoslovakia (10 Jun 1975).

Programmes have been written for fast reactors for the calculation of criticality, neutron flux spatial distribution in the core and the reflectors including other core characteristics, such as the distribution of heat generation and reactor output, reactivity coefficients, the effectivity of regulating and compensating elements, reactivity changes, and others. The programmes using a 26-group system in one-dimensional and two-dimensional geometry are based on the Galerkin variation method. Programmes have also been written based on the finite element method. The programmes have been tested in the calculation of two variants of the BN 350 fast power reactor and of the BN 600 reactor. In both cases, following the fundamental calculation of criticality and neutron fields the reactivity coefficients and some other physics parameters have been determined.

- 1897 (INIS-mf--4280, pp 137-144) EFFICIENT CALCULATION METHODS OF FAST REACTOR AND FUEL CYCLE THEREOF. Kujal, J.; Kujal, B. 1976.

(In Czech).

From Scientific and technical conference commemorating the 20. anniversary of the Nuclear Research Institute's foundation; Rez, Czechoslovakia (10 Jun 1975).

The ZELVA programme is briefly described for the calculation of a fast reactor and its fuel cycle. The programme includes a neutron-physics calculation and consequent thermal-hydraulic and strength calculations. The neutron calculation includes the determination of 26-group microscopic cross sections, the calculation of criticality and neutron flux distribution. The physics calculation is carried out one-dimensionally in a 26-group approximation. The thermal-hydraulic calculation determines the basic heat technology characteristics of the fuel element and fuel assembly and is performed for maximally loaded fuel assemblies in the core and in the breeding blanket. The maximal temperatures of the fuel pin cladding and the maximal fuel temperature are determined from the calculated values of thermal power density provided the input and output coolant temperatures are given. Also determined are friction losses, losses due to location, to spacing elements and other local losses. The strength calculation includes the determination of strength, deformation and function criteria characterizing the fuel pin and the fuel assembly claddings. It is estimated that the cladding is strained by a time-variable overpressure of fission products and a time-variable heat stress. The calculation is performed for the maximally loaded fuel pin and fuel assembly in the core and in the breeding blanket. The programme was tested by calculating a cylindrical reactor with a thermal output of 1,470 MW and a fast converter with a thermal output of 1,000 MW. Input data were taken over from projected reactors similar to the Soviet-made BN-600 and BN-350 reactors.

- 1898 (INIS-mf--4280, pp 167-169) NEUTRON TRANSPORT BY SODIUM PIPES GOING THROUGH FAST REACTOR SHIELDING 20. Burian, J.; Rataj, J. 1976. (In Czech).

From Scientific and technical conference commemorating the 20. anniversary of the Nuclear Research Institute's foundation; Rez, Czechoslovakia (10 Jun 1975).

On the WWR-S reactor, neutron transport was studied through a straight cylindrical sodium conduit 400 mm in diameter as was neutron transport through a conduit with elbows, and the effects were investigated of gaps and insulation materials on neutron transport with the purpose of determining the neutron transport through the sodium conduit in the fast reactor shielding. 7 different types of detectors were used for neutron detection, the neutron flux distribution in complex geometry was studied by the three-dimensional Boltzmann transport equation using the Monte Carlo method by means of an IBM 370/135 computer. When sodium is surrounded by a thick iron layer, neutron transport through sodium is decisive for long distance transport. If there is a gap or insulation around the conduit, neutrons passing through this medium also affect the neutron spectrum in sodium.

- 1899 (INIS-mf--4280, pp 182-189) PROGRESS IN THE CONSTRUCTION OF EXPERIMENTAL SODIUM LOOP SMT-1. Paryzek, R.; Rezac, J.; Pasek, M. 1976. (In Czech).

From Scientific and technical conference commemorating the 20. anniversary of the Nuclear Research Institute's foundation; Rez, Czechoslovakia (10 Jun 1975).

Experimental sodium loop SMT-1 being built

in the Nuclear Research Institute is designed for heat technology research of fast reactor fuel assemblies and for research and development of sodium technology. In addition to the sodium circuit proper, the equipment will consist of sodium preliminary cleaning and disposal stations, a measuring and a control unit, a 1000 kVA power supply and a ventilation system. The sodium circuit is divided into two branches operating at temperatures of 500 and 600 degC, respectively. For this purpose, an air cooler is inserted in the circuit and a contactless ejector-type heat exchanger. The circuit will also be equipped with 7 electromagnetic flowmeters without iron core, a contactless sodium level indicator and with a system of sodium leak indication.

- 1900 (INIS-mf--4280, pp 170-181)
INSTITUTE'S RESEARCH INTO THERMAL PARAMETERS OF FAST REACTOR FUEL ASSEMBLIES. Sulc, V.; Pasek, M.; Hrehor, M.; Masek, V.; Nechvatal, J. 1976. (In Czech).

From Scientific and technical conference commemorating the 20. anniversary of the Nuclear Research Institute's foundation; Rez, Czechoslovakia (10 Jun 1975).

Temperature fields in fuel assemblies modelling the BN-600 reactor assemblies for the case of deflection of the side rod from the nominal position in the direction normal to the assembly wall were studied using the experimental sodium loop S III permitting operation at sodium coolant temperature of 400 degC. The experimental assembly consisted of 16 rods, temperatures were measured with thermocouples placed on a measuring rod in a regular helical configuration with a pitch of 856 mm. The first thermocouple was placed 5 mm from the beginning of the heated length, the last at a distance of 775 mm. This configuration made it possible to measure the temperature field both along the periphery and along the length of the heating rod. The temperature profiles along the periphery of the measuring rod showed along the entire length of the rod temperature maxima in two contact places, this for all heated rods.

- 1901 (INIS-mf--4280, pp 156-162)
HYDRODYNAMICS OF EMERGENCY COOLING OF A SODIUM-COOLED FAST REACTOR. Macek, J.; Pfann, J. 1976. (In Czech).

From Scientific and technical conference commemorating the 20. anniversary of the Nuclear Research Institute's foundation; Rez, Czechoslovakia (10 Jun 1975).

A mathematical model is described permitting the quantitative study of transients in the primary coolant circuit and in the core of a sodium-cooled fast reactor in a loop configuration. The possibility of sodium boiling is not considered. The circuit is described using a system of nonlinear differential equations, the initial conditions for the solution of the system are based on the steady nominal state. The analytical solution is described of the outage of all pumps and simplifications facilitating the solution are given. A modified Runge-Kutta method permitting the automatic adjustment of the integration step length thus considerably accelerating the whole calculation, proved suitable for the integration of the nonlinear system of differential equations. The model is applicable not only for emergency after-cooling but also for other transients provided the steam phase is not formed in the coolant. The derived analytical relations facilitate good orientation in the changes in the flow rate and the pump run-out.

- 1902 (INIS-mf--4280, pp 190-196)
HYDRODYNAMICAL EXPERIMENTS ON AN ENLARGED MODEL OF A FAST REACTOR FUEL ASSEMBLY. Mantlik, F.; Hejna, J.; Cervenka, J. 1976. (In Czech).

From Scientific and technical conference commemorating the 20. anniversary of the Nuclear Research Institute's foundation; Rez, Czechoslovakia (10 Jun 1975).

The HEM-1 equipment in the Nuclear Research Institute is designed for the hydrodynamic research of fuel assemblies of fast power reactors of the BN-600 type, intended for obtaining information on hydrodynamic conditions in an infinite rod bundle with a local geometrical failure. The model consists of 19 aluminium tubes with a relative spacing of 1.17 enclosed in a hexagonal containment whose walls are in contact with the side tubes. The outer tube diameter is 120 mm, the total model length is 6 m. The central tube may be deflected towards one or two neighbouring tubes to contact. Static pressure is measured by two probes on each of four measuring rods and by 27 probes spaced evenly along the channel length by 250 mm. Static pressure measurement shows that the pressure gradient may be considered to be constant starting at a distance of 2000 mm from the inlet, i.e., about 40 hydraulic channel diameters.

- 1903 (INIS-mf--4315) REACTIONS OF STABILIZED AUSTENITIC STAINLESS STEEL IN LIQUID SODIUM. Konvicka, H.R. (Vienna Univ. (Austria)). 1977. 100p. (In German). Dep. NTIS (US Sales Only), PC A06/MF A01. TL

- 1904 (INIS-mf--4316, pp 43-47) BINARY CYCLE WITH POTASSIUM AND N_2O_4 VAPORS FOR NPP WITH A FAST SODIUM REACTOR. Popyrin, L.S.; Starostenko, V.I.; Starostenko, N.N. 1977. (In Russian).

Dissociating gases as coolants and working fluids in power plants. 3. Pt. . Collection of papers (1972-1975).

Possible ways have been investigated to increase the efficiency of NPP with fast sodium reactors. Using a binary cycle with potassium and N_2O_4 as working fluids may be one of the promising means of improving NPP characteristics. Such an approach allows the substitution of intermediate circuit with potassium for that with sodium in a three-circuit scheme. In this case, the heat-exchangers used will be much more compact as compared to water heat-exchangers. The developed mathematical model of NPP with potassium and N_2O_4 as working fluids has allowed engineering and economic investigations and a complex optimization of the plant parameters to be performed. The results obtained show that it is promising realize the above-mentioned plant and to seek for the ways of a further increase in its efficiency.

- 1905 (INIS-mf--4373, pp vp) CONTAINMENT OF THE CREYS-MELVILLE PLANT. 1977. (In French).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

CONTAINMENT SHELLS; FIRES; FUEL CANS; REACTOR CORE DISRUPTION; SODIUM; SUPER PHENIX REACTOR; TANKS

- 1906 (INIS-mf--4385) HEAT REMOVAL FROM A COMPLETELY BLOCKED BREEDER ELEMENT BY SODIUM BOILING. Schwartz, P. (Karlsruhe Univ. (TH) (Germany, F.R.)). Inst. fuer Reaktortechnik).

Jul 1977. 51p. (In German). Dep. NTIS (US Sales Only), PC A04/MF A01.

Diploma thesis.

In order to find out how a fluid flow streaming off as a film depends on the gas phase being in counterflow, two experimental apparatuses were designed and built, and measurements were made. For this purpose, the flow conditions occurring in a SNR breeder element completely blocked and boiled empty were simulated by a 31-rod bundle. A relation derived for the steady-state equilibrium between film and gas flow was used for the evaluation of the measured results. In the experiments, limiting curves could be established showing the maximum film rate flowing off as a function of the superposed gas flow rate. In addition the equilibrium parameters gas velocity, film velocity, and film thickness were determined for the material combination water/air. The influence of the spacers on the film flow was also accounted for the series of the measurements carried out. It appeared that by mounting a honeycomb spacer in the test sections the maximum film rate flowing off is drastically reduced by about 75 to 85% for the same flow rate. Mounting a second spacer, however, had no further influence on the film flowing off. It was possible to explain this phenomenon by means of the course of the experiments.

- 1907 (INIS-mf--4426, pp VII.1-20) SHORT TERMS EXPERIMENTS IN BR2 TOWARDS A BETTER FAST FUEL PERFORMANCE ANALYSIS. van Vliet, J.; Pay, A. (Societe Belge pour l'Industrie Nucleaire, Brussels). 1978.

From BR2 reactor review meeting; Mol, Belgium (1 Jun 1978).

The FARFADET experimental program to be performed in the BR2 reactor is aimed to study the begin-of-life fuel temperature with short experiments on well characterized fuel pins for LMFBR type reactors. The article reviews some aspects of the fuel physical restructuring and its chemical effects at begin-of-life, like porosity redistribution, in-pile sintering, fission gas bubble swelling, grain growth, oxygen and actinide redistribution. The planned experiments in the BR2 reactor are presented.

- 1908 (INIS-mf--4426, pp VIII.1-15) CHEMICAL INTERACTION AT THE FBR CLADDING FUEL INTERFACES. Delbrassine, A.; Retels, J.; Dirven, P. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)). 1978.

From BR2 reactor review meeting; Mol, Belgium (1 Jun 1978).

Pins containing UO_2 -30 wt.%PuO₂ and/or cesium and/or tellurium as doping elements have been irradiated for about 40 days in the BR2 reactor. The effects of two Cs/Te ratios, namely 1.3 and 4 and a wide range of D/M ratios on the inner corrosion of the clad have been investigated. The influence of Tellurium on the attack of the cladding has been pointed out. It may be responsible for the Chromium NS Nickel depletion in the grain boundaries of the steel. It is necessary to measure the effective Ts/Te ratio associated with the local corrosion layers. This local Cs/Te ratio should be more useful than the initial mean Cs/Te ratio in a pin for understanding the corrosion phenomena.

- 1909 (INIS-mf--4426, pp I.1-13) POSSIBILITIES OF APPLICATION OF EXPERIMENTAL KFK RESULTS FROM BR2 ON SNR DESIGNS. Karsten, G.; Elbel, K.; Dienst, W.; Schaefer, L. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). 1978.

From BR2 reactor review meeting; Mol, Belgium (1 Jun 1978).

A review is given of the relevant results of the technological application for the SNR300 reactor, since the BR2 reactor has been used as a test facility for the material development. Special emphasis has been laid on the fuel pin behavior under the aspect of chemical and mechanical fuel-clad interaction and on the specification of the cladding in terms of high temperature mechanical behavior in the SNR 300 reactor. A systematic analysis of urgent research topics in BR2 test facility reactor is presented.

- 1910 (INIS-mf--4426, pp II.1-13) IN-PILE INVESTIGATION OF LOCAL COOLING DISTURBANCE IN LMFBR'S (MOL 7C). Kramer, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung); Vanmassenhove, G. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)). 1978.

From BR2 reactor review meeting; Mol, Belgium (1 Jun 1978).

The main results of the MOL7C investigation program aimed at determination of consequences of local cooling disturbances in fuel pin bundles up to pin destruction are presented. The objectives of the program, the description of the test facility, a special sodium loop in the BR2-reactor, the performance of the experiments and the conclusions are described.

- 1911 (INIS-mf--4470, pp vp) MODELING OF LMFBR FUEL CLADDING MECHANICAL BEHAVIOR UNDER TRANSIENT CONDITIONS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Analytic methods must address the complexity of modeling fuel cladding response in situations where temperature, temperature gradient, strain rate, and stress are changing rapidly with time. Fuel cladding modeling is not a simple extension of the experimental and interpretative methods used in classical mechanics. New concepts are needed to describe materials in a reactor environment. Some aspects of continuum physics used to develop these new constitutive equations for fuel cladding are presented in this report. The theory of internal state variable, whose evolution is governed by ordinary differential equations, has been introduced into continuum mechanics. The components of the internal or hidden state variable vector may be scalars, components of vectors, components of second order tensors, etc. Irradiation, hardening, grain size, chemical reaction, etc., are scalar internal variables. Internal strain, dislocation loop density, etc., are all examples of second order tensors whose components are internal state variables. The internal state variables of this report are irradiation PHI which satisfies the neutron density conservation law, and hardening.

- 1912 (INIS-mf--4470, pp vp) EFFECTS OF STRESS ON SWELLING IN REACTOR FUEL CLADDING. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The purpose of this report is to describe the effect of stress on swelling in both annealed and 20% cold worked 316 stainless steel. An effect of stress on swelling in irradiated metals has been postulated for some time. Low fluence data confirmed that indeed a tensile stress can increase swelling in irradiated annealed 316 stainless steel and

that the maximum swelling occurs at an intermediate stress level which is approximately equal to the proportional elastic limit of the material. The specimens discussed above were examined by transmission electron microscopy and an effect of stress on the microstructure of the annealed and 20% cold worked 316 specimens has been observed. However, as yet, copious swelling had not occurred in the 20% cold worked material. Specimens of 20% cold worked 316 fabricated from the same heat of material as those described above have now been irradiated to sufficiently high neutron fluences that swelling has occurred in both the annealed and cold worked conditions. The specimens were 2.5 cm long, biaxially pressurized tubes, with an OD of 0.58 cm and ID of 0.51 cm. Stress levels ranged from 0 to approximately 400 MPa. Specimens were irradiated in EBR-II to maximum fluences approximately 53 dpa. Swelling in the specimens was evaluated by pre- and post-irradiation immersion density measurements.

- 1913 (INIS-mf--4470, pp vp) ANALYSIS OF CRACK PATTERNS IN FAST REACTOR FUEL PELLETS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

CRACKS;ELASTICITY;FINITE ELEMENT METHOD;FUEL PELLETS;STRESS ANALYSIS;STRESSES;SYSTEM FAILURE ANALYSIS;LMFBR TYPE REACTORS

- 1914 (INIS-mf--4470, pp vp) VARIABILITY OF CLADDING DEFORMATION OF LMFBR FUEL ELEMENTS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

CREEP;DEFORMATION;FUEL CANS;LMFBR TYPE REACTORS;PHYSICAL RADIATION EFFECTS;STAINLESS STEELS;SWELLING;SYSTEM FAILURE ANALYSIS

- 1915 (INIS-mf--4470, pp vp) STRESS-STRAIN PROPERTIES OF FAST REACTOR FUEL CLADDING UNDER CONSTANT PRESSURE TRANSIENT HEATING CONDITIONS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

FUEL CANS;FUEL PINS;HIGH TEMPERATURE;LMFBR TYPE REACTORS;STAINLESS STEEL-316;STRAINS;STRESSES;SYSTEM FAILURE ANALYSIS;TEMPERATURE DEPENDENCE;TRANSIENTS;MECHANICAL PROPERTIES

- 1916 (INIS-mf--4470, pp vp) CLADDING FAILURE BY LOCAL PLASTIC INSTABILITY. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Cladding failure is one of the major considerations in analysis of fast-reactor fuel pin behavior during hypothetical accident transients since time, location, and nature of failure govern the early post-failure material motion and reactivity feedback. Out-of-Pile transient burst tests of both irradiated and unirradiated fast-reactor cladding show that local plastic instability, or bulging, often precedes rupture. The time and location of failure is then determined by conditions favorable to bulge formation. Furthermore, the extent of the local instability limits the initial rip length and the initial area available for motion of fuel and fission products from the pin. To investigate the details of cladding bulging, a perturbation analysis of the equations governing the large deformation of a cylindrical shell has been

developed. The overall deformation history is assumed to consist of a small perturbation of the radial displacement superimposed on large axisymmetric displacements. Computations have been carried out using high temperature properties of stainless steel in conjunction with various constitutive theories, including a generalization of the Endochronic Theory of Plasticity in which both time- and time-dependent plastic strains are modeled.

- 1917 (INIS-mf--4470, pp vp) LMFBR-FUEL CLADDING THERMAL CREEP STRAIN UNDER LINEARLY INCREASING INTERNAL PRESSURE. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper describes an apparatus designed to simulate the continuously increasing internal pressure due to the accumulation of fission product gas. There is also an examination of the effects of increasing pressure on the thermal creep rate of cladding. Test specimens are 20% C.W. type 316 stainless steel tubes with 6.5 mm OD, 5.6 mm ID and 200 mm length. Increasing pressure creep tests were performed at 700°C, the hoop stress was increased in steps with 3 incremental rates of 0.15, 0.075 and 0.0375 kg/mm²h. Constant pressure creep tests were also performed at the same temperature for comparison. As mentioned above, calculations of creep behavior under increasing stress are normally made using the stationary creep rate obtained from constant pressure tests. Thus, it is important to know these creep rates agree with creep rates directly measured from strain-time plot under increasing pressure. The results show that the creep rates obtained from creep curves under increasing pressure are several times as high as the stationary creep rates measured under the same constant pressure. Creep strain rate under increasing pressure, due to the accumulation of gaseous fission products in the actual fuel cladding, is approximately 2 to 5 times that of the constant pressure creep rates, although the creep rate magnification depends on the stress and the stress-increase rate. Based on so-called time hardening theory and strain hardening theory, creep strain rate under increasing pressure is calculated using the strain-time plot under constant pressure.

- 1918 (INIS-mf--4470, pp vp) CHARACTERIZATION OF RESIDUAL STRESS PROFILES IN IRRADIATED FUEL PIN CLADDING. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Two specific applications for characterizing the residual stress in irradiated fuel pin cladding with limited, discrete data are developed in the paper. One represents a minimum effort approach and requires only four pairs of data. The end product is a 'curve' consisting of four linear segments that approximates the residual stress profile of the cladding specimen. The second technique is a semi-analytical technique for statistically fitting an assumed curve form to the data. The deformation of fueled cladding irradiation is a complex phenomenon. Modelling the deformation mechanism fuel pin cladding cannot provide detailed information fuel pin behavior without corroborative experimental data. The ability to determine the actual residual stress profile through the wall of a cladding specimen represents a new and powerful tool for evaluating cladding behavior.

- 1919 (INIS-mf--4470, pp vp) PROBABILISTIC

DESIGN METHOD FOR LMFBR FUEL RODS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Fuel rod performance analyses for design purposes are dependent upon material properties, dimensions, and loads that are statistical in nature. Conventional design practice accounts for the uncertainties in relevant parameters by designing to a 'safety factor', set so as to assure safe operation. Arbitrary assignment of these safety factors, based upon a number of 'worst case' assumptions, may result in costly over-design. Probabilistic design methods provide a systematic way to reflect the uncertainties in design parameters. PECS-III is a computer code which employs Monte Carlo techniques to generate the probability density and distribution functions for time-to-failure and cumulative damage for sealed plenum LMFBR fuel rods on a single rod or whole core basis.

1920 (INIS-mf--4470, pp vp) STATISTICAL MECHANICAL ANALYSIS OF LMFBR FUEL CLADDING TUBES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The most important design requirement on fuel pin cladding for LMFBR's is its mechanical integrity. Disruptive factors include internal pressure from mixed oxide fuel fission gas release, thermal stresses and high temperature creep, neutron-induced differential void-swelling as a source of stress in the cladding and irradiation creep of stainless steel material, corrosion by fission products. Under irradiation these load-restraining mechanisms are accentuated by stainless steel embrittlement and strength alterations. To account for the numerous uncertainties involved in the analysis by theoretical models and computer codes statistical tools are unavoidably requested, i.e. Monte Carlo simulation methods. Thanks to these techniques, uncertainties in nominal characteristics, material properties and environmental conditions can be linked up in a correct way and used for a more accurate conceptual design. First, a thermal creep damage index is set up through a sufficiently sophisticated clad physical analysis including arbitrary time dependence of power and neutron flux as well as effects of sodium temperature, burnup and steel mechanical behavior. Although this strain limit approach implies a more general but time consuming model., on the counterpart the net output is improved and e.g. clad temperature, stress and strain maxima may be easily assessed. A full spectrum of variables are statistically treated to account for their probability distributions. Creep damage probability may be obtained and can contribute to a quantitative fuel probability estimation.

1921 (INIS-mf--4470, pp vp) ANALYSIS OF FUEL/CLAD MECHANICAL INTERACTION IN NON-STEADY REACTOR OPERATION VIA FUEL ROD MODELLING CODE IANBUS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

FUEL CANS; FUEL RODS; FUEL-CLADDING INTERACTIONS; I CODES; LMFBR TYPE REACTORS; STRAINS; STRESSES; SYSTEM FAILURE ANALYSIS; COMPUTER CODES

1922 (INIS-mf--4470, pp vp) EXPERIMENTAL AND THEORETICAL EVALUATION OF DEFECT GROWTH OF FUEL PIN CLADDING BY CHEMICAL FUEL-SODIUM

INTERACTION. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In the system fuel=sodium+oxygen a reaction product (Na_2MO_4), where $\text{M}=\text{U}+\text{Pu}$, is formed after attaining an equilibrium oxygen concentration in the sodium. The reaction product is characterized by a low density (e.g. 5.5 g/cm^3) and a low thermal conductivity (about $2/10$ of that of the fuel). The melting point is about 1420°C . The reaction was predicted previously by thermodynamic assessment for UO_2 and for mixed oxide. First estimates were made for O-diffusion through the fuel to the defect after falling below the equilibrium O concentration in sodium. Because of the small diffusion coefficient, the calculated diffusion time was large and the swelling rate of the defected pin was relatively low. There are now available some results of German irradiation experiments with defected pins in sodium. The measured swelling rates are much higher than the former predicted values.

1923 (INIS-mf--4470, pp vp) 2-D THERMOELASTIC ANALYSIS OF LMFBR FUEL ROD CLADDINGS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The primary purpose of this study is twofold. Firstly, unique 2-D and 3-D temperature fields for bare finite rod bundles are presented which result from detailed distributed-parameter analyses using the slug flow model. Secondly, these results together with azimuthally varying inside and outside pressure fields are used to derive thermoelastic stress and displacement fields. The 2-D temperature fields are obtained by an iterative multicell, multiregion technique. Multicell calculations are especially important for studying the pronounced asymmetry effects induced by the coolant cells which are bounded by the bundle wall. The solution takes into account an arbitrary flow split between wall and internal subchannels as well as different heat source densities in the pins. Results indicate that the adiabatic single cell calculation is not conservative under all circumstances as has been previously assumed.

1924 (INIS-mf--4470, pp vp) MECHANICAL RESPONSE OF FFTF REFERENCE AND P1 CLADDING TUBES UNDER TRANSIENT HEATING. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

FFTF REACTOR; FUEL CANS; HIGH TEMPERATURE; MECHANICAL PROPERTIES; STAINLESS STEEL-316; STRAINS; SYSTEM FAILURE ANALYSIS; TEMPERATURE DEPENDENCE; TIME DEPENDENCE; TRANSIENTS; TUBES

1925 (INIS-mf--4470, pp vp) ANALYTICAL STUDIES OF A FINITE SIX-CHANNEL BLOCKAGE IN A LMFBR SUBASSEMBLY. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

FUEL ASSEMBLIES; HEAT TRANSFER; LMFBR TYPE REACTORS; MELTDOWN; SIMULATION; SYSTEM FAILURE ANALYSIS; THERMAL STRESSES; REACTOR CORE DISRUPTION; FLOW BLOCKAGE

1926 (INIS-mf--4492, pp v) INELASTIC RESPONSE OF METALLIC IMPACT LIMITERS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Energy mitigators on some light water reactor (LWR) spent fuel shipping casks and conceptual designs for liquid metal fast breeder reactor (LMFBR) spent fuel shipping casks rely on the inelastic behavior of metallic plate-like structures. A program was undertaken to investigate the response of mitigators to hypothetical shipping accidents with parallel experimental and theoretical studies. The theoretical studies consist of performing large deformation three dimensional transient dynamic analysis on several metal fin designs and correlating the results with the experimental results. The newly developed computer program WULFF was used. WULFF is capable of predicting the dynamic response of solids with elastic-plastic material properties. The finite element displacement formulation with the eight isoparametric elements is used with the innovation of calculating stress components at the nodes of the model. Testing of scale model fin specimens was performed at SLA's drop table facility. The test specimens are mounted on a force transducer and this assembly is bolted to the stationary bottom table. Extra weights are added to the drop table so that the total energy at impact is sufficient to cause large plastic deformation in the specimens. Accelerometers mounted on the drop table and the data from the force transducer are used to compute the energy absorbed during the test. Also framing camera coverage was used to observe the specimen deformation as it develops.

1927 (INIS-mf--4498, pp v) PARAMETRIC INVESTIGATION OF DYNAMIC FRACTURE. 1977.
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper presents results of parametric finite element fracture mechanics analyses conducted to develop an analytical model for simulating dynamic brittle fracture. Although this work may have wider general application, the dynamic loads considered here are those that may arise due to rapid release of fission gases from a failed fuel element inside a subassembly duct in an LMFBR. The analyses were conducted on an edge-cracked panel with various configurations of primary and secondary flaws similar to those investigated for quasi-static fracture earlier by Chopra, et al. The dynamic loading was in the form of a triangular pressure pulse which reached a peak value of 1000 psi in 0.1 msec and decayed to zero in 2.0 msec. The finite element mesh employed was identical to that used for previous quasi-static fracture investigations so that the dynamic and quasi-static results could be compared. For the fracture configurations analyzed, the peak dynamic crack opening stresses (COS) in the vicinity of the crack tip exceeded the static values by factor of approximately 1.5. The displacement between two opposing nodes behind the crack tip, called the crack opening displacement (COD), also was investigated. The dynamic COD was found to be approximately 1.3 times greater than the corresponding static value. This difference may be due to the different locations at which the quantities are computed.

1928 (INIS-mf--4498, pp v) TWO-DIMENSIONAL HEAT CONDUCTION CALCULATIONS ON THE PIN CLADDING AND COOLANT OF A WIRE WRAP BUNDLE. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

F CODES;FINITE ELEMENT METHOD;FUEL CANS;FUEL

CHANNELS;FUEL ELEMENT CLUSTERS;FUEL PINS;HEAT TRANSFER;LMFBR TYPE REACTORS;SPACERS;TWO-DIMENSIONAL CALCULATIONS

1929 (INIS-mf--4498, pp v) LMFBR FUEL ASSEMBLY CHANNEL WALL CROSS SECTION OPTIMIZATION. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

An LMFBR fuel assembly channel is a hexagonal thin-walled tube surrounding the fuel pin bundle to facilitate fuel handling and to control the flow of coolant through the core. The channel is subject to the internal pressure of the high temperature sodium coolant and also to a high flux of fast neutrons. This environment results in a ballooning of the channel from its original hexagonal shape due to irradiation swelling, thermal and irradiation enhanced creep, and the elastic pressure deformation of the channel material. The cross-sectional shape of the hexagonal channel is limited primarily by the principal stress and the total channel radial growth. It is presently considered preferable that adjacent channels do not touch from channel ballooning due to concerns about safety and refueling considerations. A computer program, WALLOP, has been developed as described in this paper to derive the wall shape of such an optimum channel, given realistic design and material constraints. Within WALLOP, an initial cross-sectional shape is chosen and a beam model of the wall is used to determine the elastic deflection. By assuming any changes in the axial direction along the assembly length to be gradual, a transverse slice of one wall may be reduced to a variable thickness, plain-strain, fixed-end beam with a non-planar centroidal surface and subject to transverse loading (internal coolant pressure) and axial loading (hoop force constraining the internal pressure). The deflection is combined with swelling and creep rate data on the chosen wall material and with core geometric parameters to evaluate the time for channel-to-channel contact. The computer program then sequentially perturb the wall cross section and repeats the process until channel life is maximized. This sequential search optimization process is similar to the 'Rosenbrock' algorithm.

1930 (INIS-mf--4498, pp v) COMPUTER ANALYSIS OF THE INFLUENCE OF DESIGN PARAMETERS, MATERIALS PROPERTIES AND OPERATION CONDITIONS ON THE IRRADIATION BEHAVIOR OF FAST BREEDER FUEL RODS WITH OXIDE FUEL. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Irradiation experiments have shown that clad deformations due to contact pressure have occurred starting at a certain burnup value. This threshold is determined by two processes: time to gap closure and succeeding stabilization of the outer fuel region by crack healing. Irradiation experiments and modeling analyses have revealed that the gap volume is exhausted faster than expected due to fuel swelling only. Reasons for this effect are: formation of fission gas bubbles and their migration towards the central hole with simultaneous dislocation of the fuel pellet towards the clad and/or relocation of the cracked fuel pellet, e.g. due to power changes during reactor operation. It is known, moreover, from experiments and modeling analyses that high contact pressures and resulting clad deformations occur at non-steady-state operation. Necessary conditions are: residual gap width smaller than the difference

between the thermal expansion of fuel and clad, sufficient degree of crack healing, and increase of the local rod power above the level maintained a certain time. Contact pressure and clad deformation are determined by following parameters: start and end level of the rod power, the power ramp, the mechanical stability of the fuel which is influenced by its density and crack structure and volume, and creep and/or plastic properties of fuel and clad which are governed by temperature and rod power. SATURNE-1 calculations have shown that the load onto the clad expected at power ramps depends heavily on three factors: thermal expansion of the fuel pellet (dependency on O/Me ratio, actual temperature level) healing of the peripheral radial cracks, and mechanical properties of the outer fuel region.

1931 (INIS-mf--4498, pp v) THREE-DIMENSIONAL ANALYSIS FOR FUEL PIN DEFORMATIONS IN AN LMFBR ASSEMBLY. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Based on this method, a new computational code, the Subchannel Deformation Analysis Code for Wire-Wrapped Assemblies (SHADOW), has been developed, which can be applied to seven kinds of wire-wrapped fuel assemblies with a number of fuel pins ranging from 7 to 169. This code calculates subchannel flow areas, wetted perimeters, gaps between fuel pins and distances between the outermost pins and wrapper tube wall, coupled with a thermal hydraulic code in order to get the temperature distributions of coolant and cladding in the deformed flow channels. SHADOW also calculates the magnitudes, positions and directions of contact forces loaded by the adjacent fuel pins or wrapper tube. SHADOW was utilized for analyzing the deformation due to thermal bowing of 169 fuel pins in a prototype fast breeder reactor. Conclusions derived from the present study are summarized as follows: the bowing deformations are significant throughout the overall length of the outer fuel pins and in the upper part of the gas plenum region of the inner fuel pins. These phenomena cause the axial variations of the flow areas, especially, in the outer channels and requires iterative estimations of the temperature distribution in the deformed channels. In the case of the outer fuel pins, the major contact forces are loaded near the top of each fuel pin, and in the case of the inner ones, the loads are mainly applied near the center parts of the gas plenum and core regions. These forces are not large enough to deform the cross section of the cladding.

1932 (INIS-mf--4498, pp v) RESPONSE OF A GRID SPACED FUEL ASSEMBLY TO SHIPPING ENVIRONMENT. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper presents a dynamic analysis approach for design verification of the GSA under conditions of shipping and handling using prototypical test examples of transportation by truck. Analytical response results were obtained using representative trailer bed acceleration time histories which had been recorded earlier under simulated shipping by truck transport environment. Important parameters affecting response are identified. Recognizing that the analytical estimate of the fundamental periods of the GSA cannot be very accurate owing to the complex geometry, upper bounds on acceleration responses were established using the standard response spectrum technique. Finally, the GSA design was

verified by laboratory tests—both static and dynamic. Static tests were used to calibrate displacement response of the grid spacer tabs relative to the fuel rods. Prototypical dynamic shaking table test were next employed in which the response of the GSA was determined in sinusoidal sweep type of excitation. Handling loads were represented by shock tests consisting of one-cycle impulses at various levels of acceleration. The results of laboratory testing were seen to reinforce credence of design verification by analysis of the GSA subjected to shipping and handling environment.

1933 (INIS-mf--4498, pp v) STRESS-INTENSITY FACTORS FOR IRRADIATION-EMBRITTLED HEXAGONAL SUBASSEMBLY DUCTS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The results of a finite-element analysis of the same problem are reported, and they support the applicability of beam theory to the hexcan problem. In particular, modelling the hexcan corner as a cracked strip whose crack faces are loaded with the hexcan stress distribution has been found to give very good results for the particular hexcan studied. The finite-element results for uniformly pressurized hexcans have been used as the basis for computing stress intensity factors for cracks in non-uniformly loaded hexcans, under loading conditions that might simulate those of fission gas released from failed fuel pins or other non-uniform accident loading conditions. The weight function technique employed can be used to study a variety of conditions in a quick and efficient manner, for all that is required is knowledge of the stress intensity factor of the reference problem of the uniformly-loaded hexcan and the stress distribution through the thickness of an unflawed hexcan. This information is used in conjunction with a simple representation for the crack shape, and it is shown how one may thereby compute stress intensity factors for other loading conditions by the evaluation of simple integrals. The technique is illustrated by the problem of a hexcan loaded by concentrated forces at opposite midflats. A finite-element analysis also has been carried out for the cracked hexcan loaded by midflat forces. The results of the different techniques are compared, and the simple model of a cracked strip loaded with the hexcan stress distribution is found again to agree closely with the finite-element results for the hexcan considered. However, the weight function technique is expected to apply to geometries for which the strip model is inadequate and to hold over a wider range of crack sizes.

1934 (INIS-mf--4498, pp v) MODEL FOR PRESSURE IN AN LMFBR DUCT DUE TO DISCHARGE OF GAS FROM A FAILED FUEL PIN. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In this paper an analytical model for the calculation of pressure pulses in hexagonal ducts due to discharge of gas from a failed fuel pin is developed. The analysis yields the time history of the pressure pulse which can be used in the calculation of permanent deformation of the duct or in the assessment of the susceptibility of the duct to fracture. The real physical situation of gas discharging through a pin in a duct filled with liquid is complex to model. Here the phenomenon is modeled based on some reasonable assumptions. In this model the analysis is divided into two

stages. In the first stage the gas expands as a spherical bubble, but the influence of the duct wall is taken into account. At the end of the first stage the spherical shape of the bubble is assumed to be lost and the gas is assumed to expand axially as a column. The analysis involves solving the continuity and momentum equations for the liquid along with the energy balance equation for the gas.

1935 (INIS-mf--4498, pp v) ANALYSIS OF BOWED REACTOR CORES USING THE FIAT-PROGRAM. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

FIAT is based on the NUBOW code of ANL. FIAT calculates the force equilibrium configuration of elements in a spoke, treating the ducts as beam members. The iteration technique, which is taken from NUBOW, is used in FIAT to calculate the gap distribution between the pads of the elements in the considered spoke. Knowing the gap distribution a direct solution of the unknown mechanical displacements and forces in the load planes can be given. As the gap distribution hardly changes during the burn-up history, the use of the iteration part is not required very often. The computer running times of the combined method are about 10 times less than the strict iteration technique as used in NUBOW. The computer running time reduction is improving with increasing numbers of load planes between the elements. Other improvements of FIAT over NUBOW are bowing of elements due to neutron induced swelling and creep effects, and the growth of duct width due to pressure drop and creep (bulging). In small time steps (approximately 4d) the additional amounts of swell bowing, creep bowing and growth of duct width are calculated. The equilibrium configuration is calculated after every time step to minimize the error produced by calculating at discrete time points. The calculation of the bowing reactivity of a burned core can be performed for any desired time in reactor history. The shut-down situation (low temperature level, no temperature gradients but bowed elements due to neutron fluence) can also be calculated at any time. At these time points a management of fuel elements can be prescribed, e.g. insertion of fresh elements or reshuffling of burned elements. With FIAT, the mechanical configuration of a reactor core can be calculated for the whole reactor history.

1936 (INIS-mf--4498, pp v) URANUS, A COMPUTER CODE FOR THE THERMAL AND MECHANICAL ANALYSIS OF FUEL ROD BEHAVIOR. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The URANUS-code follows a first line: a quasi 2d thermal and mechanical integral fuel rod analysis is combined with highly complex material behavior. The quasi 2d solution has been compared with a 2d-finite element solution. An excellent agreement in the axial friction force distribution was found. The mechanical description takes seven components of strains into account: elastic, time-independent plastic, creep, thermal and strains due to swelling, cracking and densification. Models for temperature distributions, heat generation, gap closure, pellet cracking, crack healing, fission gas release, corrosion, D/H distribution and Pu-redistribution are included. The URANUS-code is an advanced fuel rod design code with a variable multizone concept. Geometric nonlinearity (large displacements) are involved, the loads

(pressure or temperature) are considered as steady state or transient loads. The crack structure and the axial friction forces are calculated by means of an axial loop. The URANUS code is applicable to every fuel rod type (LWR, LMFBR, HWR). Geometric details as well as material properties are regarded as input parameters. In the paper the new basic mechanical concept, which is a superposition of id-radial solution and a id-axial solution is given as well as all basic equations. The mechanical concept determines the code structure, which also is shown in detail. Example analysis and comparisons to experiments and results from other codes are given. The result of the present work is the URANUS-code works extremely fast as calculations up to now show.

1937 (INIS-mf--4498, pp v) SUBASSEMBLY BOWING EXPERIENCE IN EBR-II. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Subassembly bowing in an LMFBR can affect core neutronics, subassembly handling, and possibly create a potential safety hazard if the phenomenon can not be characterized and predicted. Bowing may be induced by thermal gradients, differential swelling, and/or intersubassembly loading. Neutronic variations, such as long or short term power-reactivity decrement (PRD) variations, can be indicators of a bowing problem. But, because of other influences on the PRD it is very difficult to isolate the effect of bowing. As a result, subassembly bowing was not considered a problem in EBR-II until subassembly handling problems occurred. A typical handling problem, caused by top end deflection of the bowed subassembly, is centering of the subassembly removal mechanism over the subassembly. Another problem is excess loads of up to 375 kg that have been required to remove bowed subassemblies from their grid position. This latter problem is due in part, to the magnitude of bow and subassembly stiffness in addition to the overall subassembly cluster tightness caused by duct swelling and/or the occurrence of multiple bowing. Observed interferences in storing a bowed subassembly in the storage basket can be used as a qualitative gauge for bowing magnitude. In EBR-II a storage basket tube can accommodate a subassembly bow of 2.3 to 3.1 mm depending on the position in the basket. All of the bowed type 304 stainless steel subassemblies that created handling difficulties were given a post-irradiation examination. Damage that may have been caused by interference loading was found to be limited to surface scratches. The measured bow of these subassemblies range from 1.9 mm for a fueled subassembly to 5.7 mm for a neutron-source-rod thimble.

1938 (INIS-mf--4498, pp v) STRUCTURAL ANALYSIS OF EXPERIMENTAL CARBIDE FUELED DRIVER ASSEMBLY FLOW DUCT FOR TESTING IN THE FFTF. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Mixed carbide fueled driver assembly experiments will be tested in FFTF fuel driver positions as part of the National Advanced Fuel Program. The design of the experimental flow ducts must assure conformance to FFTF functional requirements in addition to service as a test vehicle for the carbide fuel irradiations. Test goals of damage fluence, burnup, and fluence to burnup ratio exceed those of the standard oxide fueled drivers. As

a consequence, the 20% cold worked type 316 stainless steel material of construction will experience significant irradiation induced creep and swelling. Additionally, the flow duct design must withstand the enhanced thermal transients produced by the action of carbide fuel during reactor scrams. A major FFTF functional requirement is that adjacent flow ducts do not touch each other except at the load pads. This requires a realistic analysis of the creep and swelling deformation of the flow duct during an experiment lifetime. Flow duct dilation, i.e., diametrical growth of the hexagonal cross section, was investigated by the MARC-CDC finite element program. Subroutines for irradiation creep and swelling were incorporated into the program. The model for the 2-D duct dilation analysis was a transverse slice through one-half of one hexagonal cross section flat (midflat to corner). The ends were free to slide outward. Generalized plane strain modelling was required to satisfy boundary conditions and high stress gradients in the corner.

1939 (INIS-mf--4498, pp v) PROOF TESTS OF IRRADIATED EBR-II DUCTS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Permanent deformation and fracture of subassembly ducts due to potential pressure pulses generated by the discharge of fission gas from fuel elements are important considerations in determining the burnup limit of fuels in LMFBR's. This paper discusses the results of a series of dynamic pressure tests that were conducted on intentionally defected irradiated EBR-II subassembly ducts. The tests were designed to simulate failure of a fuel element in a cluster of such elements. The major objective of the tests was to provide complete assurance that cladding failure of a driver-fuel element, which might conceivably cause rapid release of stored fission gas, would not deform or otherwise damage subassembly ducts in a way that would hinder movement of a control rod. Although rapid failure of a driver-fuel element in its plenum region to cause a pressure pulse is an extremely remote event, even for loss-of-flow conditions in EBR-II, the occurrence of such an event cannot be adequately disproved analytically in discussions of reactor safety. An experimental approach was therefore adopted to show -if possible- that such failure would not result in significant damage to irradiated ducts. Hindrance of movement of a control rod could then be precluded.

1940 (INIS-mf--4498, pp v) TRANSIENT NON-BOILING HEAT TRANSFER IN A FUEL ROD BUNDLE DURING ACCIDENTAL POWER EXCURSIONS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The physical problem studied is the transient non-boiling heat transfer of a cylindrical fuel rod consisting of fuel, gap, and cladding to a steady, fully developed turbulent flow. The fuel pin is assumed to be located in the interior region of a subassembly with regular triangular or square arrangements. The turbulent velocity field as well as turbulent transport properties are specified as functions of the coordinates normal to the axial flow direction. The heat generation within the fuel may be specified as an arbitrary function of the three spatial coordinates and time. A digital computer program has been developed. On the basis of finite-difference techniques, to solve the

governing partial differential equations with their associated subsidiary conditions. Results have been obtained for a series of exponential power transients of interest to safety of liquid-metal and water cooled nuclear reactors. The general physical features of transient convective heat transfer as explored by previous investigators have qualitatively been substantiated by the present analysis. Emphasis has been devoted to investigate the differences of heat-transfer (coefficient) results from multi-region analysis including a realistic fuel rod model and single-region analysis for the coolant region only. A comparison with the engineering relationships for turbulent liquid-metal cooling by Stein, which are an extension of the heat transfer coefficient concept to account for transient heat fluxes, clearly demonstrates that, at the parameters studied, Stein's approach tends to largely overestimate the convective heat transfer at early times.

1941 (INIS-mf--4498, pp vp) DEVELOPMENT OF A FINITE-STRAIN FUEL-ELEMENT MODEL. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The LIFE-III computer code has been developed to describe the steady-state performance of LMFBR fuel rods. The mechanical analysis in the original version of the code was based on the assumption that the strains in the fuel and the cladding could be considered as infinitesimal. However, this assumption is grossly violated in the vicinity of the central void, since the opening and closing of the void involves displacements of the order of the void radius. To provide a more accurate description of the mathematical behavior of the fuel element, a finite-strain fuel-element model was developed. The finite-strain calculations in the modified code are based on an incremental analysis. The deformations are assumed to be axisymmetric. The fuel element is divided into a number (≤ 10) of axial segments. Each segment is assumed to be in a state of generalized plane strain. The deformation history of the fuel element is analyzed in discrete time steps, and the basic geometry of the element is updated after each time step.

1942 (INIS-mf--4498, pp vp) DDT-A 3-DIMENSIONAL PROGRAM FOR THE ANALYSIS OF BOWED REACTOR CORES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

A 3-dimensional code was developed, to allow a good description of the whole core bowing behavior. Each single hexagonal element is described by a simple beam with 6 springs in horizontal planes where contact with other elements is possible. Each force is divided in x- and y-directions of the coordinate system. The contact forces must be in equilibrium with the mechanical displacements in both directions. Because friction forces are not included in this analysis, all forces are assumed to act normal to contacting surfaces. Gaps may open and close and contact forces may only be compressive. Because of these nonlinearities, standard mathematical techniques for solving linear structural problems are not directly applicable. So an iterative procedure is applied to achieve the equilibrium force configuration. Each iteration step corrects the largest force discrepancy between neighbouring elements. This technique requires a long computer running time to converge, but the locations of gaps and contacts normally do not change if a rough equilibrium is found. This gap distribution is

used to formulate a set of linear equations which can be solved directly. The solution is checked against the start-up gap distribution. The code can handle 300, 600, 1200 sectors of complete reactor cores. The system boundaries are symmetrically. The next step of the development is to take the friction forces into consideration which originate from the different directions of element movements. Results for a 120° sector of the KNK II reactor and for a 30°-sector of SNR 300 are available.

1943 (INIS-mf--4506, pp vp) ELASTIC AND ELASTIC-PLASTIC BUCKLING OF VESSEL HEADS COMPUTATION BY THE CEASEMT SYSTEM. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In Liquid Metal Fast Breeder Reactors of integrated type, the sodium pressure is low, but the dimensions are large. The thickness-diameter ratios are rather smaller than in other type of nuclear pressure vessels. Therefore buckling must be considered as a potential mode of failure. For instance, the bottom head of the main vessel is loaded by internal hydrostatic pressure and dead weights, and there is a need for buckling analysis. It must be pointed out that in practical cases buckling occurs after plastic deformation. Due to the effect of initial imperfections elastic buckling is rather difficult to analyse. Moreover, elastic-plastic buckling theory is not a very clear theory and consideration must be given to finite deflexions (and perhaps to initial stresses). So the choice of a suitable mathematical model is a matter of good engineering judgment. In the CEASEMT system of structural analysis by finite element method, a special program has been developed for that purpose. The model chosen (MOTAN) is near a tangent modulus model, because it seems a better approximation of the real behavior of mechanical structures. The buckling load of shells of any shape can be predicted. The method is now operational but a validation program is still in progress. Validation is especially based on buckling of vessel heads under internal pressure. Available results are given for ellipsoidal or torispherical heads. When it is possible comparison is made with known experiment.

1944 (INIS-mf--4506, pp vp) THERMAL ELASTO-PLASTIC ANALYSIS IN REACTOR VESSEL NOZZLES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In a sodium cooled nuclear fast reactor the thermal analysis has a particular importance as a consequence of the rapid transients and the high sodium conductivity. The aim of this work is to estimate the value of transient thermal stresses in elastic and plastic fields. In the first part, the loading conditions for different steady and transient operations are examined. A number of tests was planned to simulate in a mock-up the temperature changes in the upper part of the vessel for different thermal transients considered most dangerous: scram to full power, electric supply fault in a pump with reactor scram, pump seizure with reactor scram. A mock-up in scale 1 : 4 simulating the vessel area above the core was built-up. The heat transfer is simulated with a mass transfer; with this simulation it is not considered that the heat conduction in the sodium and the thermal gradients measured are higher than in the reactor. The temperature transients have been obtained with a different salt concentration: the water simulates the hot

sodium and the water salted represents the cold sodium. In the second part of the work is considered the thermo-elastic structural response using the FEM analysis. The elastic approach has had two functions: the first is to find a simplified structure in two dimensions that has the same behavior as the 3-D nozzle; the second is to follow the thermal transient to look for the time of maximum stress. For different geometry, in a previous paper, the stress distribution and the influence of some characteristic parameters have been examined. As a result of this part an axisymmetric geometry has been chosen and also the time of maxim.

1945 (INIS-mf--4506, pp vp) INFLUENCE OF SODIUM TEMPERATURE, GAMMA HEATING AND WALL THICKNESS ON DESIGN STRENGTH OF SLSF CONTAINMENT VESSELS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

LMFBR TYPE REACTORS;PRESSURE VESSELS;STEELS; STRESS ANALYSIS;SYSTEM FAILURE ANALYSIS; TEMPERATURE DEPENDENCE;THERMAL SHOCK;THICKNESS; REACTOR VESSELS

1946 (INIS-mf--4506, pp vp) THERMAL RATCHETTING IN PRESSURE VESSELS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

During its lifetime a nuclear power plant undergoes cyclic thermal loadings, i.e. time variations of temperature spatial distribution; these thermal cycles, combined with mechanical loads (pressure), can produce dimensional changes for creep, plastic cycling and/or ratchetting. This problem is in special way important for a Liquid Metal Fast Breeder Reactor (LMFBR) because temperatures in it are high (about 500°C) and change either in radial direction or in axial direction. While the plastic behavior of a structure subjected to mechanical loads and/or mechanical cyclic loads can be well studied with the by Finite Element analyses and several different computer programs, the thermal plastic behavior was studied for simplified conditions only. Actually, the Bree, Miller, Burgreen model considers a thin cylindrical shell with a radial thermal gradient and in uniaxial stress state while Kainins, Updike extend this model to a biaxial stress state. To obtain numerical results of the thermo-plastic behavior of cylindrical shells, a mathematical model was developed and the special purpose computer program PLAST was implemented.

1947 (INIS-mf--4506, pp vp) ELASTOPLASTIC AND NONSYMMETRIC BUCKLING ANALYSIS BY FINITE ELEMENT OF SUPER-PHENIX PRIMARY VESSEL HEAD. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

CLOSURES;DEFORMATION;ELASTICITY;FINITE ELEMENT METHOD;PLASTICITY;PRESSURE VESSELS; STEELS;STRESS ANALYSIS;STRESSES;SUPER PHENIX REACTOR

1948 (INIS-mf--4513) ANNUAL REPORT 1977. (Stichting Energieonderzoek Centrum Nederland, The Hague). Jun 1978. 92p. (In Dutch). Dep. NTIS (US Sales Only), PC A05/MF A01.

The change of statute made in 1976 has been followed by a shift of emphasis towards non-nuclear subjects in the ECN Research Centre. A performance testing of experimental wind

turbines to be installed at the Petten site, is done in the framework of the national energy R and D programmes. An effort on technical superconductors and superconducting coils became the subject of a cooperation with the Netherlands Applied Research Organisation TNO and some industrial companies. Programmes on magnetohydrodynamic conversion and on energy storage in flywheels are in advanced preparation. A large experimental effort is devoted to the fast reactor development programme of Germany, Belgium and the Netherlands. Complete is the estimation of the reactivity effects of fission products in the prototype fast reactor under construction at Kalkar. The HFR will also be used in a new series of transient overpower experiments on fast reactor fuel pins. ECN activities in the nuclear fuel cycle are concentrated on the disposal of radioactive work in salt domes.

- 1949 (INIS-mf--4525, pp 5-6) DESIGN AND PRODUCTION TECHNOLOGY OF THE ISLIN 5/700 ELECTROMAGNETIC PUMP. Alekseev, R.A.; Bezgachev, E.A.; Golovanov, M.M.; Ivanov, V.V.; Karasev, B.G. 1978. (In Russian).

From 9. Riga meeting on magnetic hydrodynamics; Riga, USSR (17 Apr 1978).
BOR-60 REACTOR; ELECTROMAGNETIC PUMPS; FABRICATION; HIGH TEMPERATURE; INDUCTION; SODIUM; SPECIFICATIONS

- 1950 (INIS-mf--4525, pp 3-4) RESULTS OF EXPERIMENTAL RESEARCHES OF ELECTROMAGNETIC PUMPS FOR THE BOR-60 REACTOR. Andreev, A.M.; Karasev, B.G.; Kirillov, I.R.; Ogorodnikov, A.P.; Ostapenko, V.P.; Semikov, G.T. 1978. (In Russian).

From 9. Riga meeting on magnetic hydrodynamics; Riga, USSR (17 Apr 1978).
BOR-60 REACTOR; EFFICIENCY; ELECTROMAGNETIC PUMPS; HIGH TEMPERATURE; INDUCTION; PERFORMANCE TESTING; SODIUM

- 1951 (INIS-mf--4525, pp 7-8) DESIGN AND OPERATING EXPERIENCE OF ELECTROMAGNETIC PUMPS IN THE PRIMARY AND SECONDARY CIRCUITS OF THE BR-10 FACILITY. Andreev, A.M.; Glukhikh, V.A.; Ivanov, V.V.; Karasev, B.G.; Ostapenko, V.P.; Semikov, G.T.; Kulikov, Yu.L.; Letov, B.G.; Nikulin, M.P.; Timosh, B.S. 1978. (In Russian).

From 9. Riga meeting on magnetic hydrodynamics; Riga, USSR (17 Apr 1978).
ELECTROMAGNETIC PUMPS; HIGH TEMPERATURE; INDUCTION; OPERATION; PERFORMANCE; SBR-5 REACTOR; SODIUM

- 1952 (INIS-mf--4541, pp vp) SIMPLE STEEL/WATER MODEL FOR PRELIMINARY STUDIES OF ACOUSTIC VIBRATION IN LMFBR. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

One source of vibration excitation in Liquid Metal Fast Breeder Reactors is the acoustic energy which is generated by the circulating pump and transmitted through the fluid to various structural components. Since most of the energy occurs at fairly low frequencies, that of low harmonics of blade passing frequency, only the very large components have resonant frequencies such that they are significantly excited. To gain some preliminary understanding of the extent and magnitude of vibration in fast reactors therefore, a simple model has been constructed in which only the major components are represented. The modelling theory is discussed and it is shown that adequate representation of the steel/sodium

reactor materials can be obtained in the model based on the use of steel/water. The model represents a pool design with a primary tank of 3 1/4 metres diameter and typical components scaled in proportion; however, it does not necessarily relate to any specific reactor design. The pump acoustic source is represented by an underwater loudspeaker system and vibration amplitudes are scaled according to typical pressures generated by reactor circulators. Results from the model include calibration data for the acoustic source and measurements of acoustic pressure throughout the primary flow circuit and the inner and outer pools. Stresses are measured on structural components over a frequency range scaled from reactor frequencies and compensated for the characteristics of the acoustic source. Appreciable stresses are found on all the components in the primary circuit, not necessarily only those close to the simulated pump source. After scaling them to reactor size and allowing for the source calibration, it is found that stresses are unlikely to be sufficiently high to cause damage.

- 1953 (INIS-mf--4541, pp vp) FRACTURE OF LONGITUDINALLY CRACKED DUCTILE TUBES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
CRACKS; DUCTILITY; FAILURE MODE ANALYSIS; LMFBR TYPE REACTORS; TUBES

- 1954 (INIS-mf--4541, pp vp) TEMPERATURE AND STRESS RESPONSES OF CLEAN EVAPORATOR TUBES UNDER OSCILLATORY WATER RIVULET ACTION IN DNB REGION. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

DEPARTURE NUCLEATE BOILING; EVAPORATORS; FINITE ELEMENT METHOD; HEAT EXCHANGERS; HEAT TRANSFER; LMFBR TYPE REACTORS; MATHEMATICAL MODELS; STRESS ANALYSIS; TUBES

- 1955 (INIS-mf--4541, pp vp) VIBRATIONAL ANALYSIS OF THE SUPERPHENIX INTERNAL SHELLS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

ANNULAR SPACE; DAMPING; FLUCTUATIONS; HEAT EXCHANGERS; HYDRODYNAMIC MASS EFFECT; MECHANICAL VIBRATIONS; OSCILLATION MODES; PUMPS; RESONANCE; SHELLS; SINGULARITY; SUPER PHENIX REACTOR

- 1956 (INIS-mf--4541, pp vp) COMPUTERIZED THERMAL INPUT FOR STEAM GENERATOR STRESS ANALYSIS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In order to perform structural analyses of the CRBRP steam generators (hydraulic transients) detailed thermal profiles must be developed and transmitted to the stress analyst in a usable form. A unique, semi-automated computer system has been developed to optimally accomplish this task. The transient data are given in the specification in the form of sodium and water (steam) flow, temperature and pressure at the inlet of the steam generator as a function of time. The steam generator is divided into fourteen distinct zones each consisting of several subzones. Masses of thermal data must be generated to provide temperature distributions in each subzone as a function of time. A simplified nodal model of the entire steam generator is used to translate

the specification transient data into boundary conditions for the detailed subzone models. This is accomplished by the SETS (Superheater-Evaporator Transient Simulation) program. A one-to-one correlation between the stress and thermal models has been set up to enhance accuracy and simplify data transmittal. The stress analyst develops a finite element model to the degree of detail required by the MARC computer code. Nodes are added to this model and the thermal model development is accomplished using STAMP (Stress to Thermal Adaption Model Program). Thermal analysis is accomplished using a finite difference code - TAP (Thermal Analyzer Program). The SETS tape is used to provide the boundary conditions. Significant temperatures and temperature differences are output, isotherms are also output TEMPOS (Temperature Postprocessor) is then used to extract the thermal data and present the results in such a manner that the maximum stress intensity range may be calculated.

1957 (INIS-mf--4541, pp vp) FLOW-INDUCED VIBRATION BENCH TESTING OF REACTOR COMPONENTS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

CLINCH RIVER BREEDER REACTOR; DAMPING; HYDRODYNAMIC MASS EFFECT; MECHANICAL VIBRATIONS; REACTOR COMPONENTS; SCALING LAWS; STRUCTURAL MODELS

1958 (INIS-mf--4541, pp vp) EFFECT OF TUBE-SUPPORT INTERACTION ON THE DYNAMIC RESPONSE OF HEAT EXCHANGER TUBES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

As a part of ANL's Flow-Induced Vibration Program in support of the Clinch River Breeder Reactor Plant (CRBRP) steam generator design activity, tube/support interaction experiments are being performed not only to gain the insight into the dynamic behavior of CRBRP steam generator tubes, but also to provide the basis for developing design guidance. In the experiment, the tube and supports are prototypic, the tube spans nearly prototypic, and overall length approximately one-third that of a straight tube segment of the CRBRP steam generator. The test model features a 15.88 mm dia. x 3.18 mm thick x 5.08 m long tube (2-1/4% Cr - 1% Mo) surrounded by water, a 101.60 mm thick tubesheet, seven 25.44 mm thick stainless steel supports with 0.254 mm tube-to-support hole clearances and two closely-spaced supports adjacent to the excitation span. The tube is excited by electromagnetic coils simulating crossflow excitation; two displacement transducers, orthogonal to one another, are positioned at the center of each span. Tube response of the midspan locations were measured as a function of various input excitation levels and frequencies. Results were compared with analytical results based on multi-span tube with 'knife-edge' supports at the support locations. Test results show that the resonant frequencies are close to those of the analytical results; experimental modes are in good agreement with analytical predictions or the tube/support configuration tested, peak displacement amplitudes are always lower than those of analytical results; closely-spaced double supports do not effectively dampen out the vibration responses of the adjacent spans and significant orbital motions exist in the spans away from the excitation. It is concluded that a mathematical model assuming a multi-span beam fixed at the tubesheet with intermediate

'knife-edge' supports satisfactorily predicts natural frequencies and mode shapes for the tube/support configuration studied.

1959 (INIS-mf--4543, pp vp) THIN CIRCULAR CYLINDER UNDER AXISYMMETRICAL THERMAL AND MECHANICAL LOADING. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

To assess structural integrity of components subjected to cyclic thermal loadings one must look at thermal ratchetting as a possible failure mode. Considering a thin circular cylinder subjected to constant internal pressure and cyclically varying thermal gradient through the thickness Bree, J. Strain Analysis 2 (1967) No.3, obtained a diagram that serves as a foundation for many design rules (e.g.: ASME code). The upper part of the french LMFBR main vessel is subjected to an axisymmetrical axial thermal loading and an axial load (own weight). Operation of the reactor leads to cyclic variations of the axial thermal loading. The question that arises is whether or not the Bree diagram is realistic for such loading conditions. A special purpose computer code (Ratch) was developed to analyse a thin circular cylinder subjected to axisymmetrical mechanical and thermal loadings. The Mendelson's approach of this problem is followed. Classical Kirchhoff-Love hypothesis of thin shells is used and a state of plane stress is assumed. Space integrations are performed by Gaussian quadrature in the axial direction and by Simpson's one third rule throughout the thickness. Thermoelastic-plastic constitutive equations are solved with an implicit scheme (Nguyen). Thermovisco-plastic constitutive equations are solved with an explicit time integration scheme (Treanor's algorithm especially fitted). A Bree type diagram is obtained for an axial step of temperature which varies cyclically and a sustained constant axial load. The material behavior is assumed perfectly plastic and creep effect is not considered. Results show that the domain where no ratchetting occurs is reduced when compared with the domain predicted by the Bree diagram.

1960 (INIS-mf--4543, pp vp) PLASTIC CREEP BEHAVIOR ANALYSIS OF AN ELBOW PIPE. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

High temperature components are liable to strong creep effects, in LMFBR for example. This phenomenon is analyzed here for an elbow pipe. Comparisons are performed between experiments and computations using CEASEMT codes, particularly piping-system analysis code TEDEL. The reference is an experiment from Batelle's Columbus Laboratories and concerns a 4 in-sh 10 elbow pipe assembly made of type 304 stainless steel at room temperature and then at 593°C (1100°F) during more than 300 hours. The assembly is loaded by inplane moment; displacements, temperatures and strain measurements are picked up. Computations are first performed using the finite element codes developed by CEA. For creep analysis iterative processes referring to the normality principle of HILL are used: during steps of iterations, equilibrium and yield criterion are always met.

1961 (INIS-mf--4543, pp vp) CREEP-FATIGUE DESIGN CRITERIA FOR APPLICATION TO A 2 1/4 CR.-1 MO LMFBR STEAM GENERATOR. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

A study was undertaken to evaluate alternate criteria for creep fatigue evaluation of LMFBR steam generators constructed of 2 1/4 Cr.-1 Mo alloy steel. A review was conducted of the ASME Criteria for Design of Elevated Temperature Class 1 Components and the existing data available for 2 1/4 Cr.-1 Mo alloy steel. The current method in the Code for calculating creep-fatigue interaction damage using linear summation of fatigue damage and creep damage was employed on the existing test data. Alternately, the strain range partitioning method was considered. It was determined that the Code method is generally conservative for predicting life of the 2 1/4 Cr.-1 Mo test specimens, and the strain range partitioning method, as originally proposed, appeared to be unconservative when extrapolated to the low strain range long hold time service required by the design. A series of creep relaxation analyses employing the linear time fraction creep damage rule to evaluate the effect of plant thermal transients on the predicted design life and sensitivity of calculated creep damage to various material behavior parameters was conducted. Several creep equations and the effect of biaxiality were studied to determine their effect on stress relaxation and calculated creep damage. Calculated stress relaxation results from various creep equations were compared to measured relaxation from structural tests to aid in selecting an appropriate creep equation. In order to explore the sensitivity of calculated creep-fatigue damage to primary stress intensity, an infinite thick-cylinder was analyzed for variable pressure conditions using elastic-plastic-creep finite element techniques. In order to evaluate the effect of potential elastic follow-up effects on calculated creep damage, a built-in cylinder was also analyzed by finite element techniques.

1962 (INIS-mf--4543, pp vp) BEHAVIOR OF ANNEALED TYPE 316 STAINLESS STEEL UNDER MONOTONIC AND CYCLIC BIAXIAL LOADING AT ROOM TEMPERATURE. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

ANNEALING;ELASTICITY;LMFBR TYPE REACTORS; PLASTICITY;REACTOR MATERIALS;STAINLESS STEEL-316;STRAINS;STRESSES

1963 (INIS-mf--4543, pp vp) INSTABILITY ANALYSIS OF ELBOWS IN THE PLASTIC RANGE. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This work presents some basic considerations underlying the instability in the plastic range of typical LMFBR elbows (radius-to-thickness ratio $r/t=20-50$), and the impact of these considerations on the practical aspects of solution by computer methods. The elbows are made from a strain hardening material and they are loaded by in-plane closing or opening bending moments. A simplified method of computer analysis is used and the particular type of instability considered is that of geometric buckling into a mode that is characterized by significant ovalization of the cross section and by a relatively gradual variation in shape along the length of the elbow. Emphasis is placed on the effects of including geometric nonlinearities (large displacement terms) in the inelastic (time-independent) analysis. Numerical results are presented for a typical LMFBR elbow. A general discussion of the basic concepts and definition of the instability of structures made from

strain hardening or from elastic materials is given. Such a discussion is needed to facilitate a complete understanding of the present numerical results and to avoid possible misinterpretations in a comparison of these and previous results. The major conclusions of this investigation are as follows: (1) Geometric nonlinear effects are significant, (2) The inclusion of such effects in an analysis of an elbow subjected to either a closing or an opening moment reveals that the buckling load is lower for the case of a closing moment. The buckling load for this case corresponds to a maximum point (zero slope) in a typical load-displacement curve, and (3) the omission of geometric nonlinear effects in the inelastic analysis leads to a load-displacement curve in which the load monotonically increases with displacement.

1964 (INIS-mf--4543, pp vp) CREEP BUCKLING OF SHELLS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Because of the characteristics of LMFBR primary piping components (thin-walled, low pressure, high temperature), the designer must guard against creep buckling as a potential failure mode for certain critical regions, such as elbows, where structural flexibility and inelastic response may combine to concentrate deformation and cause instability. The ASME Boiler and Pressure Vessel Code, through its elevated temperature Code Case 1592 (Section III, Division 1) provides design rules for Class 1 components aimed at preventing creep buckling during the design life. A similar set of rules is being developed for Class 2 and 3 components at this time. One of the original concepts behind the creep buckling rules was that the variability in creep properties (especially due to the effects of prior heat treatment), the uncertainty about initial imperfections, and the lack of confirmed accuracy of design analysis meant that conservatism would be difficult to assure. As a result, a factor of ten on service life was required (i.e. analysis must show that, under service conditions that extrapolate the life of the component by ten times, creep buckling does not occur). Two obvious problems with this approach are that: first, the creep behavior must also be extrapolated (since most creep experiments are terminated at a small fraction of the design life, extrapolation of creep data is already an issue, irrespective of the creep buckling question); second the nonlinear creep analysis, which is very nearly prohibitively expensive for design life histograms, becomes even more costly. Analytical results for an aluminum cylindrical shell subjected to axial loads at elevated temperatures are used to examine the supposed equivalence of two types of time-dependent buckling safety factors - a factor of ten on service life and a factor of 1.5 on loading.

1965 (INIS-mf--4543, pp vp) ANALYSIS OF PIPING SYSTEMS FOR CREEP. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The purpose of this paper is to provide a survey of the current situation and to attempt to explain and clarify a number of misconceptions which have arisen in connection with approximative techniques. Recently there have become available several Finite Element computer codes for the analysis of vessel type structures which can deal with any geometry, loading condition and material behavior.

However, a finite element analysis of a large diameter complex piping system for high temperature reactor service is simply not feasible economically to the detail and accuracy of a pressure vessel analysis; in any case, the aim is to produce a flexible system. Even those simplified finite element methods which have been designed to cope principally with piping are not practical for the initial stages of design for elevated temperature. Ideally, a simple flexibility beam approach, similar to those classically used in elastic analysis, would be desirable. Such an approach has so far only been achieved using simplifications to the creep response. The two simplified methods which are primarily used are known generally as Kachanov's Method and the Method of Superposition of States. In this paper the reliability of these approximations is examined for simple piping components under combined loading; these also serve to clarify a number of misconceptions which have arisen over the methods. It is then discussed not only how these simplified methods can be adapted to the analysis of piping systems but also how a more general beam type approach could be constructed, and what sort of information would be necessary for their successful implementation.

1966 (INIS-mf==4543, pp vp) EVALUATION OF INELASTIC ANALYSIS METHODS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The purpose of this presentation is to explore this question relative to interim inelastic analysis guidelines that have been recommended for use in the design of liquid-metal-cooled reactor system components. Interim constitutive equations have been developed for describing the inelastic behaviors of types 304 and 316 stainless steel and 2 1/4 Cr-1 Mo steel. Although differing in detail, the recommended constitutive equations for these three materials are basically of the same form. They rest on the fundamental assumption that an increment of total strain consists of four additive contributions: thermal, elastic, plastic, and creep. The specification of the plastic and creep strains follows from the concepts of classical kinematic hardening plasticity theory and strain-hardening creep theory. These concepts are augmented, however, with ad hoc rules to account for such things as cyclic hardening and/or softening, aging, and the effects of prior inelastic straining on subsequent plastic and creep deformations. Uniaxial and multiaxial exploratory tests were used to identify the essential inelastic behavioral features of significance, and these features, in turn, provided the basis for choosing the interim constitutive equations. Typical exploratory test results are presented to support the basic constitutive equation framework and to illustrate how the ad hoc rules were developed, evaluated, and improved. To assess the validity of the constitutive equations and analysis methods for high-temperature design, benchmark structural tests were conducted under time-varying loadings at elevated temperatures, and the results were compared to inelastic analysis predictions. Representative test results for beam and plate structures and for a pipe subjected to repeated thermal shocks are presented and compared to predictions.

1967 (INIS-mf==4543, pp vp) FINITE ELEMENT ELASTIC PLASTIC ANALYSIS OF LMFBR COMPONENTS. 1977.

From 4. international conference on

structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

ELASTICITY;FINITE ELEMENT METHOD;H CODES; LMFBR TYPE REACTORS;MECHANICAL STRUCTURES; PLASTICITY;REACTOR COMPONENTS;RESPONSE FUNCTIONS;SHELLS;STRESS ANALYSIS

1968 (INIS-mf==4543, pp vp) FATIGUE DAMAGE ANALYSIS PROCEDURE FOR THERMAL LINERS SUBJECTED TO RANDOM THERMAL FLUCTUATIONS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Of particular concern in this paper is the structural analysis of thermal shock liners in liquid sodium systems subjected to sinusoidal and random thermal fluctuations produced by mixing of hot and cold fluid streams. The development of the response function, relating thermally induced surface stresses to both low and high frequency sinusoidal surface temperature fluctuations, enables one to develop a method of determining fatigue damage of the structural components by conventional random vibration analysis techniques. Having the response function for the thermal shock liners enables one to predict the fatigue damage due to thermal fluctuation having either wide-band or narrow-band spectral characteristics. Although results are presented for a particular analysis - the analytical procedures can be easily adapted to consider other thermal environments.

1969 (INIS-mf==4543, pp vp) SOME COMMENTS ON THERMOMECHANICAL CONSTITUTIVE EQUATIONS FOR INELASTIC ANALYSIS OF LMFBR COMPONENTS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The safety analysis of the primary and secondary circuits of sodium-cooled fast breeder reactors (LMFBR) requires systematic methods to predict the inelastic behaviors of metal structures under complex thermomechanical loading conditions. To provide those methods, constitutive equations have been recommended for interim use. This paper is concerned with the physical evidence and consistency of these constitutive equations. In particular, shortcomings of the recommended constitutive theories may result from the bilinear approximation of the stress-strain-curve, the kinematic linear hardening rule and the very special kind of the proposed creep laws. Moreover, the elastic-plastic and creep equations are formulated independently: except for hardening due to prior creep strains, possible interactions between viscoelastic and plastic material behaviors are neglected. Finally, the temperature appears as a given parameter, that means. The recommended constitutive assumptions do not allow for any coupling effects between the deformation and temperature field. A consistent set of constitutive equations should be based on a correct evaluation of the second law of thermodynamics. Looking at the specific mathematical form of the recommended constitutive theories, one can generalize the classical elastic-plastic and creep equations. However, in a more systematic approach to the problem of inelastic analysis, general results of the modern theories of non-linear material behavior should be utilized. These results provide systematic methods to establish constitutive equations which are thermomechanically consistent as well as experimentally practicable.

1970 (INIS-mf==4592) DYNAMIC RESPONSE OF

SINGLE HEXAGONAL LMFBR CORE SUBASSEMBLY WRAPPERS. Jaeger, T.A.; Boley, B.A. (comps.). (Commission of the European Communities, Brussels (Belgium)). 1977. vp. (CONF-770807--(Pt.E)).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

To analyze the dynamic structural response of the LMFBR core subassembly hexagonal wrappers to postulated local energy releases and the sensitivity of the response to variations in both the pressure loading and the material properties of the stainless steel, a finite-element computer code STRAW has been developed. A series of experiments was performed to study the effects of variations in material properties. The amount of coldworking to which the Type 316 stainless steel is subjected has a strong influence upon the ductility and the elastic yield point. The usual fabrication process produced a nominally 20% coldworking with a yield point of about 680 MPa. By designing a special set of dies for the drawing process, a very low ductility hexcan was produced for which the yield point was raised to 820 MPa. Conversely, the yield point was lowered to 170 MPa by a solution annealing process producing a highly ductile test hexcan. A metallurgical study was conducted to find a representative brittle simulant material for the irradiated end-of-life steel properties. An aging treatment for Type 446 stainless steel was developed which reproduced the expected tensile-flow behavior of the in-pile subassembly. Further study is underway to investigate the fracture properties of the simulant material. The pressure pulses were generated by the controlled expansion of high-pressure detonation products from low-density explosives detonated inside a vented steel canister. The orifice configuration of the canister and the charge mixture ratio were designed to produce two specified pulse shapes. A charge containing 37.7 g PETN mixed with 35 wt % inert, hollow-glass microballoons developed a pressure pulse peak of 9.5 MPa at 1.0 ms. Increasing the PETN to 41 g resulted in a 14.6 MPa peak pressure, and increasing the explosive concentration to 90 wt % in the mixture increased the burning rate and the pulse risetime, so that the peak occurred at 0.6 ms.

1971 (INIS-mf--4592) DYNAMIC LOADINGS OF SODIUM-WATER REACTIONS IN LMFBR AND FUSION POWER DESIGNS. Jaeger, T.A.; Boley, B.A. (comps.). (Commission of the European Communities, Brussels (Belgium)). 1977. vp. (CONF-770807--(Pt.E)).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In liquid metal fast breeder reactor and lithium cooled fusion reactor, a sodium loop is being proposed to transfer heat from the primary coolant loop to the steam turbine cycle. Although by careful design and quality assurance programs, the probability for steam generator tube failure can be minimized, failure will still occur. The direct contact of sodium and water would cause a chemical reaction where hydrogen and sodium compounds are produced. This paper presents an evaluation of the potential hazards as a result of such a reaction. An analytical method is developed to investigate the extent of the reaction zone and the propagation of the pressure wave in the sodium system. In the calculation, the chemical reaction is assumed to be instantaneous, governed by the equation $2\text{Na}(l) + \text{H}_2\text{O}(l) \rightarrow \text{Na}_2\text{O}(l) + \text{H}_2(g) + 31.4 \text{ K cal/gm. mole}$. Both the temperature and pressure rise in

the reaction zone can be established from the energy balance and the equation of state for the gaseous product. As a consequence of the energy released, the chemical products suddenly expand with a high velocity. The expansion also generates a shock wave in both the water and the sodium systems. Results indicate that the reaction zone can expand in a rate of 1500 ft/sec and a shock wave with initial strength of 2300 atmospheres propagates with a speed of 8000 ft/sec into the sodium system. The propagating characteristics of the shock wave are obtained by solving the basic fluid equations. The shock wave decays rapidly, in the neighborhood of milliseconds, as soon as the reaction zone stops to expand. The decrease in the reaction zone pressure allows more water to react with the sodium and a second pulse is generated.

1972 (INIS-mf--4592, pp vp) EFFECT OF HIGH EXTERNAL THERMAL LOADING ON LMFBR SUBASSEMBLY INTEGRITY. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

FUEL ASSEMBLIES; HEXAGONAL CONFIGURATION; LMFBR TYPE REACTORS; LOSS OF COOLANT; THERMAL STRESSES

1973 (INIS-mf--4592, pp vp) ANALYSIS OF EBR-II STEAM-GENERATOR SYSTEM FOR PRESSURE PULSES DUE TO LEAKS OF STEAM OR WATER INTO SODIUM. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

EBR-2 REACTOR; LEAKS; LIQUID METALS; REACTOR ACCIDENTS; SODIUM; STEAM GENERATORS; WATER; PRESSURE GRADIENTS; LEAKS

1974 (INIS-mf--4592, pp vp) ANALYSES OF HYDRODYNAMIC EFFECTS OF LARGE SODIUM-WATER REACTIONS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Large leak sodium-water reactions that would occur in a steam generator of LMFBR causes abrupt changes of pressure and velocity of fluid in a secondary sodium system and relief system. This paper describes SOWACS-III together with its model and method. Results of analyses are also given, the comparison with experimental results of initial pressure spike being included. SOWACS-III treats the system which consists of the steam generator, vessel, valve, pump and pipe, and uses the following models and methods. (1) Components are assumed to be one-dimensional. (2) Pressure wave propagation near a reaction zone, where hydrogen is generated, is analyzed with the spherical co-ordinate (sphere-cylinder model). (3) A moving boundary is formed by contact of sodium with other fluid such as hydrogen and nitrogen. The boundary travels without mixing of sodium and another fluid through the boundary (boundary tracking model). The boundary can be treated not to move from the original place (fixed boundary model). (4) Pressure wave propagation is analyzed by the explicit method of characteristics in one-dimensional Eulerian co-ordinate. (5) Flow-induced force is analyzed by momentum balance. (6) The lateral motion of relief piping caused by the force is analyzed by NASTRAN code. Analyses were carried out for large sodium-water reaction experiments in SWAT-3 rig of PNC by using the sphere-cylinder model. The calculated pressure spike in the reaction vessel was compared with the measured one for a

few milliseconds after water injection. The calculated value and measured one were 6.4 ata and 6.7 ata for peak pressure and 0.6 ms and 2.8 ms for rising time, respectively.

- 1975 (INIS-mf--4592) THERMAL AND STRESS ANALYSES OF MELTDOWN CUPS FOR LMFBR SAFETY EXPERIMENTS USING SLSF IN-REACTOR LOOPS. Jaeger, T.A.; Boley, B.A. (comps.). (Commission of the European Communities, Brussels (Belgium)). 1977. vp. (CONF-770807--(Pt.E)).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The test trains for the Sodium Loop Safety Facility (SLSF) in-reactor experiments, which simulate hypothetical LMFBR accidents, have a meltdown cup to protect the primary containment from the effects of molten materials. Thermal and stress analyses were performed on the cup which is designed to contain 3.6 kg of molten fuel and 2.4 kg of molten steel. The cup principal components are: 1. A 38 mm diameter tungsten spike which provides initial fuel quenching and prevents fuel boiling, 2. A 73 mm inside diameter tungsten liner to isolate the support vessel from the molten material high initial temperature, 3. An insulator which is an expedient for extending the experiment time, and 4. An Inconel 625 vessel which provides the structural support to withstand the thermal and pressure stresses. The spike, liner, and insulator are supported by a hemispherical tungsten end cap which fits inside the hemispherical bottom of the support vessel. This vessel is attached to the 316 stainless steel test train with an Inconel 750 wire-formed retaining ring. Thermal analyses were performed with the Argonne-modified version of the general heat transfer code IHTB, based on the instantaneous addition of 3200°K molten fuel with a decay heat of 9 W/gm and 1920°K molten steel. These analyses have shown that the cup will adequately cool the molten materials. The maximum temperature occurs at the center of the fuel region but it is always less than the fuel boiling point. The maximum temperature occurs at the center of the fuel region but it is always less than the fuel boiling point. The most severe heating occurs when there is no sodium flow outside the cup. For this case the sodium boils (approximately 1200°K) and the Inconel vessel and tungsten liner temperatures are approximately 1250°K and 2420°K, respectively.

- 1976 (INIS-mf--4592, pp vp) BULK COOLANT CAVITATION IN LMFBR CONTAINMENT LOADING FOLLOWING A WHOLE-CORE EXPLOSION. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

An LMFBR core undergoing an explosion transmits energy to the containment in a series of pressure waves and the containment loading is determined by their cumulative effect. Pressure waves are modified by their interaction with the coolant through which they propagate. In liquids in their normal state the main effect of the interaction is to steepen the fronts of waves leading to shock formation but in liquids which have undergone bulk cavitation the interaction is much stronger and is expected to be dissipative and dispersive. Since reflections of initial pressure waves from the core at free surfaces lead to the establishment of a state of tension and consequently of bulk cavitation over large volumes of the coolant, it is necessary to model both the induction of bulk cavitation by tension waves and the interaction of pressure

waves with cavitating liquid in realistic containment loading calculations. The progress which has been achieved in such modelling and first indications for the effect of bulk coolant cavitation in LMFBR containment loading. A model describing the interaction between cavitation bubbles and a host liquid subject to time-varying pressure fields, including the physical process of momentum conservation, phase change, heat conduction and mass diffusion is presented and used to determine which of the various physical processes involved dominate the results. It is shown that if only interested in bulk effects momentum transfer is the chief factor for the pressures and timescales of interest. To determine the effect of the state of purity of the liquid, the main characterising parameters are the initial radii of nucleation sites and their number density, the latter being very influential. To identify the important differences between model water and reactor sodium as far as cavitation is concerned are chiefly the lower surface tension and higher concentration of initially larger nucleation sites in the sodium.

- 1977 (INIS-mf--4592, pp vp) VAPOR TRANSPORT AND PRESSURE RISE BEHIND THE LINER OF REACTOR CONTAINMENTS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

CONTAINMENT; LINERS; LMFBR TYPE REACTORS; PRESTRESSED CONCRETE; REACTOR ACCIDENTS; STEELS; PRESSURE GRADIENTS

- 1978 (INIS-mf--4592, pp vp) STRUCTURAL RESPONSE OF LARGE LMFBR HEAD CLOSURES TO HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

A finite-element model was developed to assess the ability of a pool-type LMFBR head closure to sustain an energetic HCDA. From a conceptual representation of an expanded EBR-II type deck, it was determined that there is a reasonable amount of symmetry in the structure that an initial model would consist of a 120° sector. This model accounts for the main structural elements of the closure head: beams, plates, inner ring, in-tank component nozzle and concrete shielding. The mass of the in-tank component was treated as a concentrated mass located at its attachment to the structural members. The ANL-developed SADCAT code was used in this study. The code is an explicit, three-dimensional finite-element code developed for analyzing transient, nonlinear, structural problems. The code has a triangular plate element and a three-dimensional continuum element. The above model was used to study the effect of several design variations on the structural response of the deck to an energetic HCDA. Two types of outer boundary conditions were considered to determine their effect on the closure head's displacements and stress levels. In addition to the above study, an analysis was conducted to determine the structural condition of the concrete shielding. Based upon the above preliminary calculations of pool-type LMFBR head closures, it is indicated that the shield-deck can be designed to survive the given HCDA loads without failure.

- 1979 (INIS-mf--4592, pp vp) RESPONSE OF FAST REACTOR PRIMARY CONTAINMENTS TO WHOLE CORE ACCIDENT LOADINGS. 1977.

From 4. international conference on

structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

COMPUTER CODES; CONTAINMENT; HYDRODYNAMICS; LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION; RESPONSE FUNCTIONS

- 1980 (INIS-mf--4592, pp vp) STRUCTURAL DYNAMICS STUDIES FOR SINGLE AND CLUSTERED SNR-300 FUEL ELEMENTS: A COMPARISON OF ANALYTICAL AND EXPERIMENTAL RESULTS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Structural damage may occur in liquid metal cooled fast breeder reactors (LMFBR) when local failures escalate to a fast release of excessive thermal energy within one fuel element. Time scales (5. to 50. msec) and peaks of associated pressure pulses (50. to 500. bar) may exceed the burst pressure of the hexagonal wrapper containing the pin-bundle. Though after fuel element rupture the pressure level in the vicinity is markedly reduced, still a significant pressure field is propagating through the core. Therefore surrounding subassemblies or control rods are exposed to impulsive differential pressures causing duct flexure as well as local cross section deformation. A theoretical and experimental research program at GfK Karlsruhe investigates the mechanical effects of conservative pressure transients on LMFBR fuel elements. As part of the program this paper describes two computer codes (BEDYN-2, CORE-1) for non-linear structural dynamics of single as well as clustered subassemblies. Important features and idealizations of mechanical models are discussed before the codes are applied to several loading cases. Then two types of subassembly deformation experiments are outlined, and original records of the pressure pulses are used as input for BEDYN-2 to predict and compare deformation histories with experimental findings: (1) Quasistatic plane strain flat-to-flat loading of the hexagonal wrapper with and without the stiffening pin-bundle. (2) Transverse impulsive loading of a complete 1:1 SNR-300 type fuel element model on roller supports. In addition, pressure records from underwater explosion tests on SNR code models are taken as input for the multirow dynamics code CORE-1 and fairly good correlations were found between measured and predicted permanent deformations.

- 1981 (INIS-mf--4592, pp vp) ENERGY SOURCE FOR MODELING HCDA BUBBLES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This work describes the development and characterization of an energy source that models the assumed initial conditions and the subsequent expansion of an HCDA bubble in a simple /30-scale model of the Clinch River Breeder Reactor (CRBR). The source is used in a study of the phenomena associated with expanding, rising HCDA bubbles, the transport of radioactive core materials to the cover gas region, and the loading resulting from the impact of the coolant slug with the vessel cover. The source uses high-temperature, high-pressure water as the bubble material. The water, initially in the vessel's lower core, is released by opening an explosively driven, fast-opening valve between the lower and upper core barrel. When released, the water reaches a superheated state and flashes to vapor. The vapor expands out of the lower core, through the upper core barrel, and into the vessel containing the coolant simulant (water). High-speed photography is used to observe the

expansion through the transparent walls of the vessel. Characterization tests were performed with and without a vessel cover and with both nitrogen gas (1500 psi, room temperature) and water (1160 psi, 563°F, 0% quality) as the bubble material. The surface motion, bubble volume, pressures in the lower and upper core, and the slug impact pressure were measured. From these, the displaced volume-time, entrainment-time, pressure-volume change, gas-work-volume change, and slug kinetic energy-volume change relations were derived.

- 1982 (INIS-mf--4592, pp vp) SIMULATION STUDY OF THE SECONDARY RUPTURE OF SG TUBING FOR LMFBR. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper, which is a part of a large research program for safety design of SG, intends to present and discuss the results of a simulation study made to determine the water injection rate and the tube fracture aspect as a function of the wastage shape. In the experiments, pre-milled aluminium tubes were allowed to burst under internal pressure given by water at room temperature. Aluminium was used because the mechanical properties were similar to those of the steel under the actual service conditions. The wastage was typified by two kinds of artificial surface flaw: one the flat bottom groove milled along the length of the tube (the flat flaw) and the other the oval concave milled by running a cylinder grinder bar transverse to the tube axis (the cylindrical flaw), the length (flat flaw) and the radius (cylindrical flaw) as well as the thickness remaining underneath the flaw being the variables. The inner diameter of the test tube was always 19.4 mm, while the wall thickness was so varied between 1.2 and 2.6 mm that the underflaw wall be selectable between 0.6 and 1.6 mm for the burst pressure to be between 80 and 200 kg/cm². To determine the water injection rate, the pressure history was measured by semi-conductor transducers at two points near the flaw; and the progress of crack opening was recorded by stroboscopic photography. The water injection rates calculated on those two measurements will be presented. It will be shown that the time the injection rate became constant was several tenths of a msec for the flat flaws and several msec for the cylindrical flaws, and that the time the crack opened to 90% of the maximum was again several tenths of a msec for the former and several msec for the latter.

- 1983 (INIS-mf--4592, pp v) HYDRODYNAMIC AND ELASTOPLASTIC STRUCTURAL ANALYSIS OF FAST BREEDER REACTOR CORE ACCIDENT. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper describes the principles and examples of applications of an explicit Lagrangian coupled finite difference-finite element code HEMP-ESI developed in order to calculate the structural consequences of hypothetical core disruptive accidents (HCDA) in nuclear reactors. The explicit solution algorithm of the finite difference scheme used to discretize the hydrodynamic fluid domains is shown to be very similar to that used for the solution of the finite element discretized shell structures, hence permitting an easy and efficient coupling. Two examples of simulation show the applicability of the method to nuclear reactor core safety analysis (test problem).

1984 (INIS-mf--4592, pp v) DETAILED NON-LINEAR ANALYSIS OF THE RESPONSE OF THE DECK OF A FAST BREEDER REACTOR UNDER ACCIDENTAL SODIUM IMPACT. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

CONCRETES;CONTAINMENT;EXPLOSIONS;FINITE ELEMENT METHOD;PLATES;RESPONSE FUNCTIONS;SODIUM;STEELS;SUPER PHENIX REACTOR;REACTOR ACCIDENTS;IMPACT SHOCK;THERMAL STRESSES;THREE-DIMENSIONAL CALCULATIONS

1985 (INIS-mf--4623) EXAMPLE OF REGULATING PROCESS FOR AN ADVANCED REACTOR: CREYS-MALVILLE. Cravero, M. (Electricite de France, 92 - Courbevoie. Service Etudes et Projets Thermiques et Nucleaires). 1978. 18p. (CONF-780584--4). Dep. NTIS (US Sales Only), PC A02/MF A01.

From International conference on regulating nuclear energy; Brussels, Belgium (16 May 1978).

The general philosophy for the regulatory control of the Super-Phenix fast breeder reactor was the same as for other types of nuclear power plants. The licensing process for both conventional and nuclear aspects was also identical to that for other nuclear units. On the other hand, safety recommendations were especially prepared by the CEA's appropriate services on the basis of acquired experience. The paper analyses the development of the licensing process for the reactor and knowledge to be gained for future units of this type.

1986 (INIS-mf--4638, pp vp) PRESSURE TRANSIENT ANALYSIS IN SINGLE AND TWO-PHASE WATER BY FINITE DIFFERENCE METHODS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

An important consideration in the design of LMFBR steam generators is the possibility of leakage from a steam generator water tube. The ensuing sodium/water reaction will be largely controlled by the amount of water available at the leak site, thus analysis methods treating this event must have the capability of accurately modeling pressure transients through all states of water occurring in a steam generator, whether single or two-phase. The equation systems of the present model consist of the conservation equations together with an equation of state for one-dimensional homogeneous flow. These equations are then solved using finite difference techniques with phase considerations and non-equilibrium effects being treated through the equation of state. The basis for water property computation is Keenan's 'fundamental equation of state' which is applicable to single-phase water at pressures less than 1000 bars and temperatures less than 1300°C. This provides formulations allowing computation of any water property to any desired precision. Two-phase properties are constructed from values on the saturation line. The use of formulations permits the direct calculation of any thermodynamic property (or property derivative) to great precision while requiring very little computer storage, but does involve considerable computation time. For this reason an optional calculation scheme based on the method of 'transfinite interpolation' is included to give rapid computation in selected regions with decreased precision. The conservation equations were solved using the second order Lax-Wendroff scheme which includes wall friction, allows the formation of shocks and locally supersonic flow. Computational boundary conditions were found from a method-of-characteristics solution

at the reservoir and receiver ends. The local characteristics were used to interpolate data from inside the pipe to the boundary.

1987 (INIS-mf--4638, pp vp) DAMPING OF THE RADIAL IMPULSIVE MOTION OF LMFBR CORE COMPONENTS SEPARATED BY FLUID SQUEEZE FILMS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The core deformation of a liquid metal cooled fast breeder reactor (LMFBR) due to local pressure propagation from rapid energy releases is a complex three-dimensional fluid-structure-interaction problem. High pressure transients of short duration cause structural deformation of the closely spaced fuel elements, which are surrounded by the flowing coolant. Corresponding relative displacements give rise to a squeezing fluid motion in the thin layers between the subassemblies. Therefore significant backpressures are produced and the resulting time and space dependent fluid forces are acting on the structure as additional non-conservative external loads. Realistic LMFBR safety analyses of several clustered fuel elements have to account for such flow induced forces. This paper describes two fluid flow models (model A, model B), which are shown to be suitable for physically coupled fluid-structure analyses. Important assumptions are discussed in both cases and basic equations are derived for one- and two-dimensional incompressible flow fields. The interface of corresponding computer codes FLUF (model A) and FLOWAX (model B) with structural dynamics programs is outlined. Finally fluid-structure interaction problems relevant to LMFBR design are analyzed; parametric studies indicate a significant cushioning effect, energy dissipation and a strongly nonlinear as well as time-dependent damping of the structural response.

1988 (INIS-mf--4638, pp vp) PTA-1 COMPUTER PROGRAM FOR TREATING PRESSURE TRANSIENTS IN HYDRAULIC NETWORKS INCLUDING THE EFFECT OF PIPE PLASTICITY. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Pressure pulses resulting from a sodium-water reaction occurring in a steam generator in an LMFBR intermediate heat transport system may plastically deform the thin-walled piping typically used in such systems. This plastic deformation has a significant effect on pressure transient propagation, since it limits the peak pressure transmitted out of a pipe to approximately its yield pressure if the pipe is sufficiently long. The computer code PTA-1 incorporates the effect of plastic deformation of the piping on pressure transient propagation in complex hydraulic networks. Although it was developed for predicting the propagation of pressure pulses in the intermediate heat transport system of an LMFBR, it may also be used to analyze transient propagation in other complicated piping networks for which fluid-hammer theory is appropriate. PTA-1 uses the one-dimensional method-of-characteristics applied to fluid-hammer analysis of pressure transients in large piping systems. Non-linear convective terms, pipe friction and fluid compressibility are included in the formulation of the governing equations. The effect of plastic deformation of piping is incorporated through a modified fluid wave speed which varies with deformation history at each computational node in a plastically deforming pipe. An iterative procedure is employed to

locate the characteristic curves, through the wave speed, are a function of fluid pressure and pipe deformation history. Various types of junctions and fittings may be specified; these include closed ends, multi-branched tees, surge tanks, sudden expansions and contractions, dummy junctions, acoustic-impedance discontinuities, non-reflecting far-end boundaries, and simple models for pumps and rupture disks.

- 1989 (INIS-mf--4690, pp v) DYNAMIC INSTABILITY ANALYSIS OF AXISYMMETRIC SHELLS BY FINITE ELEMENT METHOD WITH CONVEXED COORDINATES. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

A rectilinear shell element formulated in the convected (co-rotational) coordinates is used to investigate the effects of edge conditions on the behaviors of thin shells of revolution under suddenly applied uniform loading. The equivalent generalized nodal forces under uniform loading are computed to the third order of the length of each element. A dynamic buckling load is defined as the load at which a great change in the response is observed for a small change in the loading. The problem studied is a shallow spherical cap. The cap is discretized into a finite number of elements. This discretization introduces some initial imperfections into the shell model. Nonetheless, the effect of this artificial imperfection is isolated from the effect of the edge conditions provided the same number of elements is used in all the cases. Four different edge conditions for the cap are used. These boundary conditions are fixed edge, hinged edge, roller edge and free edge.

- 1990 (INIS-mf--4690, pp v) SOME TRENDS IN CONSTITUTIVE EQUATION MODEL DEVELOPMENT FOR HIGH-TEMPERATURE BEHAVIOR OF FAST-REACTOR STRUCTURAL ALLOYS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

CREEP; ELASTICITY; EQUATIONS; HIGH TEMPERATURE; LMFBR TYPE REACTORS; MATHEMATICAL MODELS; MECHANICAL PROPERTIES; PLASTICITY; REACTOR MATERIALS; STAINLESS STEEL-304; STAINLESS STEEL-316

- 1991 (INIS-mf--4690, pp v) PROBABILISTIC ASSESSMENT IN STRUCTURAL ANALYSIS OF MAIN COOLANT SYSTEMS FOR LMFBR UNDER ANTICIPATED THERMAL TRANSIENTS. 1977.

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

LMFBR TYPE REACTORS; PRIMARY COOLANT CIRCUITS; PROBABILITY; RELIABILITY; RISK ASSESSMENT; SECONDARY COOLANT CIRCUITS; THERMAL STRESSES; TRANSIENTS

- 1992 (INIS-mf--4760, pp 59-98) EXPLOSIVE PLUGGING OF NUCLEAR HEAT EXCHANGERS. Crossland, B.; Bahrani, A.S.; Townsley, W.J. (Queen's Univ., Belfast, Northern Ireland (UK)). 1976.

From 3. international symposium on the use of explosive energy in manufacturing metallic materials of new properties by explosive welding, cladding, hardening, and metal-powder pressing processes; Mariánské Lázně, Czechoslovakia (19 Oct 1976).

The explosive plugging techniques developed in the Department of Mechanical and Industrial Engineering of the Queen's University of

Belfast for the reheater and superheater of the PFR reactor and for the reheater of the AGR type reactor are described. For the PFR a point charge system was used which causes a spherical expansion of the plug, which gives two zones of welding. Using a parallel stand-off welding set up, it proved difficult or impossible to avoid a crevice. Consequently, a rim charge set up was developed which gives a circular ring expansion of the plug with two zones of welding. The problem is also considered of the distortion of holes adjoining the hole to which a plug is welded. Bunting of adjoining holes in order to reduce the distortion was also investigated. The described equipment is shown in drawings.

- 1993 (INIS-mf--4773) CASSANDRE: A 2D MULTIGROUP DIFFUSION BENCHMARK CODE FOR FAST REACTORS POWER EXCURSIONS ANALYSIS. Bardiaux, M.; Devooght, J.; Mund, E.; Saragossi, I. (Brussels Univ. (Belgium)); Dugnoille, P.; Goldwasser, M.; Joly, J.; Siebertz, A. (Brussels Univ. (Belgium)); Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)). 1978. 13p. (CONF-781022--64). Dep. NTIS (US Sales Only), PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

CASSANDRE uses a four group 2D diffusion model with finite elements up to quintic order. Time dependence involves a generalization of the quasistatic approximation with a different amplitude function for each group. The coupled shape and amplitude equations are cast into integral form and solved iteratively using a Newton-SOR algorithm. A simplified thermohydraulics feedback is incorporated. A great number of parametric studies involving different time steps and number of Newton iterations show that power transients accuracy near prompt critical may be strongly affected by reactivity and shape function convergence.

- 1994 (INIS-mf--4816, pp 2-6) IMPORTANCE OF THE FAST BREEDER FOR OUR ENERGY SUPPLY. Faude, D. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). 1978. (In German).

From Information meeting on questions concerning nuclear energy; Karlsruhe, F.R. Germany (13 Oct 1977).

A fuel balance is made for LWR and FBR reactors. The natural uranium consumption for different reactor strategies is compared, and it is found that the fast breeder technology is the only technical possibility of energy supply that is not hampered by limited fuel reserves.

- 1995 (INIS-mf--4816, pp 11-16) SAFETY OF SODIUM-COOLED FAST BREEDERS, USING THE EXAMPLE OF THE 300 MWE BREEDER PROTOTYPE POWER PLANT AT KALKAR (SNR 300). Heusener, G. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). 1978. (In German).

From Information meeting on questions concerning nuclear energy; Karlsruhe, F.R. Germany (13 Oct 1977).

The danger potential of fast breeders is similar to that of other reactor systems; suitable safety measures are required to prevent accidents. As in all reactors, the safety measures taken in fast breeders are highly reliable. There are measures to prevent accidents (preventive measures) and safety measures to keep the consequences of accidents small and under control within the reactor system (containment measures). The SNR 300 is constructed so as to control even the most extreme possible accident, the Bethe-Tait

accident which is highly improbable, within the containment so that no excessive radiation pollution of the environment will occur.

- 1996 (INIS-mf--4816, pp 7-10) TECHNICAL CONCEPT OF FAST BREEDER POWER PLANTS. Hueper, R. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueker). 1978. (In German).

From Information meeting on questions concerning nuclear energy; Karlsruhe, F.R. Germany (13 Oct 1977).

Fast breeders generate power and at the same time, produce nuclear fuel. The author describes constructional principles and the state of the art of sodium-cooled fast breeders.

- 1997 (INIS-mf--4816, pp 17-21) LARGE-SCALE USE OF SODIUM. Peppler, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). 1978. (In German).

From Information meeting on questions concerning nuclear energy; Karlsruhe, F.R. Germany (13 Oct 1977).

The author outlines reserves, chemical and thermodynamic properties of sodium, as well as its use in breeder technology where heat transfer components play a special role with regard to their behaviour during accidents in case of sodium-air and sodium-water reactions. Sodium fires in the primary Na systems can be excluded as these spaces are always flooded with nitrogen. In the secondary system an effective sodium catcher system prevents sodium fires; Na-H₂O reactions in case of a steam generator break are prevented by a reliable, quick-action failure relief system.

- 1998 (INIS-mf--4857) FIVE YEARS EXPERIENCE OF THE OPERATION OF THE SODIUM CIRCUITS OF THE DOUNREAY PROTOTYPE FAST REACTOR. Walford, J.G. (UKAEA Dounreay Nuclear Power Development Establishment). 1978. 12p. Dep. NTIS (US Sales Only), PC A02/MF A01.

During the five years since the various coolant circuits of the PFR fast breeder were filled, some 36,000 hours of operation of each of the main plant components have been accumulated. A description is given of experiences on the major sodium pumps and their ancillaries, the sodium pipework and valves, the 'operational' in-reactor mechanisms such as control rod drives and the intermediate heat exchanger valves. The performance of sodium instrumentation is discussed as is that of the sodium purification loops. The major successes, as well as a number of operating difficulties and the measures taken or proposed to overcome them, are described.

- 1999 (INIS-mf--4862) SAFETY STUDY. THE CLINCH RIVER BREEDER REACTOR PLANT. Piper, H.B.; Leaver, D.E.; Wood, P.J. (Project Management Corp., Oak Ridge, TN (USA)). 1978. 9p. (CONF-781022--84). Dep. NTIS (US Sales Only), PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

A risk assessment has been performed to provide a realistic evaluation of the risk to the public from the Clinch River Breeder Reactor Plant (CRBRP), to place that risk in perspective relative to other societal risks, and to provide a basis for assessing the comparability of the risk from the CRBRP with risks associated with previously licensed US reactors. The assessment formally and systematically evaluated the CRBRP risk and

showed that the risk associated with postulated accidents of the CRBRP is small when compared with non-nuclear risks to which the local population is already exposed, and that CRBRP risks are comparable to those from light-water reactors (LWR's) of the current generation as characterized in the Rasmussen Study.

- 2000 (INIS-mf--4911) INVESTIGATIONS OF TEMPERATURE FLUCTUATIONS IN LIQUID SODIUM IN ORDER TO DETERMINE CHARACTERISTIC FLOW PARAMETERS AND THERMOCOUPLE TRANSFER FUNCTIONS. Benkert, J. (Hannover Univ. (Germany, F.R.). Fakultät fuer Maschinenwesen). 1977. 104p. (In German). Dep. NTIS (US Sales Only), PC A06/MF A01.

Diss. (D.Eng.).

Because of narrow geometries and high thermal loads mainly thermocouples are used to monitor the coolant state in the reactor core, the heat exchangers and the piping. The aim of the present paper is to determine the transfer behavior of different types of thermocouples, the local influencing of the temperature fluctuations and to what extent the heat transfer coefficient fluid/wall may be evaluated from the measurements performed. The experiments to study these problems were carried out in three test beds for liquid sodium. The good agreement between model calculations and the experiments made allow the experimenter to make interpretations of extreme flow conditions on one hand and the owners of plants to select and apply optimally thermocouples under normal operating conditions on the other.

- 2001 (INIS-mf--4960) SODIUM FIRE AND CONTAINMENT LEAKAGE. Person, L.W. (Argonne National Lab., IL (USA)). 1978. 3p. (CONF-781022--108). Dep. NTIS (US Sales Only), PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

A combination of statistical response surface method and existing sodium fire computer modelling will provide functional requirements to the reactor containment designer and give the range of radioactivity release to the environment via containment leakage for the radiation health physicist. It can be generalized as a viable approach to reducing the time and cost required to obtain estimates of radiation safety aspects of nuclear power reactors.

- 2002 (INIS-mf--4974, pp 334-344) TEMPERATURE DEPENDENCE OF VOID PARAMETERS IN THE OKH16N15M3BR STEEL FUEL CANS UNDER RADIATION CONDITIONS. Ostrovskii, Z.E.; Prokhorov, V.I. 1978. (In Russian). From Conference on reactor materials science; Alushta, Ukrainian SSR (29 May 1978).

An attempt is made to explain various swelling of the BOR-60 reactor fuel cans irradiated in the same fuel packet under identical conditions. Studies on the effect of different factors as possible causes of dispersions in the can swelling of identical fuel elements have shown that the most important factor is various gas content in the OKh16N15M3BR steel fuel can in the process of void formation.

- 2003 (INIS-mf--5083) UK FAST REACTOR COMPONENTS. SODIUM REMOVAL DECONTAMINATION AND REQUALIFICATION. Donaldson, D.M.; Bray, J.A.; Newson, I.H. (UKAEA Risley Nuclear Power Development Establishment). 1978. 26p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Extensive experience gained at the U.K.A.E.A. Dounreay Nuclear Power Development Establishment is being applied to form the basis of the plant to be provided for sodium removal, decontamination, and requalification of components in future commercial fast reactors. In the first part of a three part paper, the factors to be taken into account, showing the UK philosophy and approach to maintenance and repair operations are discussed. In the second part, PFR facilities for sodium removal and decontamination are described and some examples are given of cleaning components such as pumps, charge machine, cold trap baskets, and steam generator units. Similar facilities at DFR are briefly described. In the third part of the paper a short description is given of the Harwell mass transfer loop, currently used to study the deposition of activated stainless steel corrosion products. Decontamination method for pipework specimens cut from the loop are described and results of first screening tests of various chemical decontaminants are presented.

- 2024 (INIS-mf--5107) COMPARISON OF SAFETY CHARACTERISTICS OF THE ADVANCED REACTOR LINES FAST BREEDER AND HTR. REPORT TO THE COMMITTEE FOR INTERNAL AFFAIRS OF THE GERMAN BUNDESTAG. MATERIALS. 1978. vp.p. (In German).

The following points are compared for SNR-300 and THTR-300: 1) Safety aspects; 2) safety characteristics, e.g. activity inventory and its enclosure by fuel can and fuel matrix, primary system and containment; 3) securing of barriers like shutdown ability, removal of afterheat, retention systems, and containment sealing; 4) emission of gaseous substances during normal operation; 5) accidents and their radiological effects.

- 2005 (INIS-mf--5114) REPORT OF THE ACTIVITIES DURING THE YEARS 1977 AND 1978. (Nijverheidsorganisatie TNO, The Hague (Netherlands). Projectgroep Kernenergie TNO). 8 Feb 1979. 28p. (In Dutch). Dep. NTIS (US Sales Only), PC A03/MF A01.

An introduction with general aspects is followed by more detailed descriptions of the separate projects concerned with sodium cooled, fast breeder reactors. These include material for use at high temperatures, safety related applied mechanics, specific sodium technology, joining techniques, heat transfer and current, not destructive testing, and pump developments.

- 2006 (INIS-mf--5244) ANNUAL REPORT 1978. (Stichting Energieonderzoek Centrum Nederland, The Hague). Jun 1979. 80p. (In Dutch). Dep. NTIS (US Sales Only), PC A05/MF A01.

The activities of ECN in 1978 give first the general analysis of the Dutch energy economy carried out by the Energy Study Centre (ESC). Studies on energy modelling and energy conservation in buildings are briefly discussed.

- 2007 (INIS-mf--5308, pp 99-117) BEHAVIOUR AND SOME PROPERTIES OF U-PU OXIDE FUEL RODS OF FAST REACTORS. Men'shikova, T.S.; Golovnin, I.S.; Flatkina, V.P.; Sytov, L.I.; Korostin, O.S.; Davydov, E.F.; Kuz'min, V.I. 1978. (In Russian).

From Conference on reactor materials science; Alushta, Ukrainian SSR (29 May 1978).

Methods of preparing homogeneous UPuO₂ specimens of different initial powders (mechanically mixed and co-precipitated) are considered. The relationships are presented

between creep rates of UPuO₂ rods at different temperatures and plutonium content. It is established that the creep rate increases with plutonium content from 10 to 30 mass.%. The dependence of the thermal resistance of a 'fuel-clad' gap on the gap width has been investigated. Thermal resistance approximately doubles with an increase of the gap from 0 to 0.4 mm. The interaction between a UPuO₂ rod and OKh16N15M3B steel clad has been studied in the BOR-60 reactor under the operational conditions.

- 2008 (INIS-mf--5308, pp 142-163) RESULTS OF IRRADIATION TESTS OF FAST REACTOR OXIDE FUEL AND FUEL ELEMENTS. Tsykanov, V.A.; Davydov, E.F.; Gryazev, V.M. 1978. (In Russian). From Conference on reactor materials science; Alushta, Ukrainian SSR (29 May 1978).

The experimental results are considered obtained in post-reactor investigations into the structure variation, plutonium redistribution, swelling and gas release in pellet and vibrocompacted oxide fuel. The data are generalized on the operating capabilities of BOR-60 fuel elements and BN-600 experimental fuel elements of UO₂ and UO₂-PuO₂ as well as on the swelling, mechanical properties and corrosion of the OKh16N15M3B steel claddings.

- 2009 (INKA-Conf--79-003-006) OXYGEN DISTRIBUTION IN FAST REACTOR OXIDE FUELS. Ewart, F.T. (UKAEA, Harwell. Chemical Technology Div.); Mari, C.M. (Milan Univ. (Italy). Istituto di Elettrochimica e Metallurgia); Fourcaudot, S.; Matzke, H.; Manes, L.; Toci, F. (Commission of the European Communities, Karlsruhe (Germany, F.R.). European Inst. for Transuranium Elements). (UKAEA Atomic Energy Research Establishment, Harwell). 1979. 5p. (IAEA-SM--236/7(preprint); CONF-790111--31). Dep. NTIS (US Sales Only), PC A02/MF A01.

From International symposium on thermodynamics of nuclear materials; Julich, F.R. Germany (29 Jan 1979).

The chemistry of irradiated fuel pins has been examined with respect to oxygen distribution. Two hypostoichiometric mixed oxide fuel pins have been irradiated to high burn up under controlled conditions in a materials testing reactor. The radial oxygen profile at the end of irradiation has been measured. Small specimens were extracted from differing points on the radius of the fuel; their oxygen potential was measured in a micro galvanic cell. The results have shown that, although the pins had initially widely different oxygen contents (O/M values of 1.948 and 1.976), the final O/M values were virtually identical at 1.992 and 1.997. This is contrary to the behaviour predicted from existing fuel chemistry theory. Oxidation above 1.997 O/M in the higher initial O/M pin is prevented by the oxidation of the clad.

- 2010 (INKA-Conf--79-082-001) INTERPRETATION OF LOCAL BOILING BEHAVIOUR IN THE KNS BLOCKAGE EXPERIMENTS. Clare, A.J. (Karlsruhe Univ. (TH) (Germany, F.R.). Inst. fuer Reaktortechnik). 1979. 1p. (CONF-7905160--1). NTIS (US Sales Only), PC A02/MF A01.

From Seminar on reactor technology; Karlsruhe, F.R. Germany (10 May 1979). BOILING; DRYOUT; FLOW BLOCKAGE; FUEL ELEMENT FAILURE; LMFBR TYPE REACTORS; MOCKUP; OSCILLATIONS; SIMULATION; SODIUM

- 2011 (INKA-Conf--79-134-001) BOPSI - A MODEL FOR THE THEORETICAL DESCRIPTION OF THE

BEHAVIOUR OF BOILING FUEL-STEEL ARRAYS SEALED OFF ON ALL SIDES. Buchner, H. (Karlsruhe Univ. (TH) (Germany, F.R.). Inst. fuer Reaktortechnik; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). (Karlsruhe Univ. (TH) (Germany, F.R.). Inst. fuer Reaktortechnik; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). 1979. 1p. (In German). Dep. NTIS (US Sales Only), PC A02/MF A01.

B CODES;LMFBR TYPE REACTORS;MELTDOWN; NUMERICAL SOLUTION;PHASE TRANSFORMATIONS;POOL BOILING;NUCLEAR FUELS;STEELS;PRESSURE DEPENDENCE;TEMPERATURE DEPENDENCE; THERMODYNAMICS

- 2012 (INTAT--78-70, pp 84-101) DESIGN VIEW POINTS AND MATERIAL ASPECTS FOR SNR ABSORBER. Duenner, P.; Toebbe, H. Aug 1978. (In German).

From KITG special meeting of the Fachgruppe Brennelemente; Bergisch Gladbach, F.R. Germany (25 Nov 1977).

The present meetings report gives a survey on the state of development for absorbers of fast reactors. Numerous materials, material combinations and geometrical configurations are for neutron-physical reasons possible for the various reactor concepts of control and shutdown. Boron carbide is primarily used as reference material in the SNR, however, numerous alternatives to realize long-term project targets are still being pursued.

- 2013 (INTAT--78.147) FAST BREEDERS AND ENVIRONMENT. Hoerning, H. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). Dec 1978. 28p. (In German). (CONF-780917i--1; INKA-Conf--78-271-003). Dep. NTIS (US Sales Only), PC A03/MF A01.

From Conference on the role of the fast breeder reactor; Essen, F.R. Germany (27 Sep 1978).

The waste heat of a fast breeder has a thermal efficiency of 40%. This is the same as in conventional power plants, i.e. not typical of nuclear power plants. Wet and dry cooling towers are built with increasing frequency in order to protect rivers. The potential danger due to high radioactivity inventories in all fuel cycle stages is not the same as its actual danger. Radioactivity sources in fast breeders are listed and compared with those of LWR reactors. Activity sources and inventories are similar. Exposure pathways are listed which might endanger the population in the vicinity of fast breeders and other nuclear facilities in case of radioactivity release. In addition to the rated values stated in the Radiation Protection Ordinance for normal operation and accidents the very low radiation doses calculated for the SNR-300 are given. This is also proved by a comparison with natural and man-made radiation and their variation in West Germany. Fuel element reprocessing, experience so far, and waste conditioning are dealt with. The toxicity of high-activity waste of fast breeders and LWR reactors is compared, and its reduction on storage is shown. The West German concept for final storage is outlined.

- 2014 (IS-T--887) ELECTRO- AND THERMOTRANSPORT OF CARBON IN IRON-NICKEL ALLOYS. Okafor, I.C.I. (Ames Lab., IA (USA)). Mar 1980. Contract W-7405-ENG-82. 104p. Dep. NTIS, PC A06/MF A01.

Results of studies on the electrotransport, thermotransport and chemical diffusion of

carbon in γ -phase iron = 32.5 wt % nickel alloys are described. The transport parameters, Z, Q and D were measured as a function of temperature and carbon concentration and values for the activation energies for diffusion and electrotransport were obtained. The solubility limit of carbon in γ -Fe-32.5 wt % Ni for the temperature range 850 to 1200°C and the electrical resistivity versus temperature for carbon concentrations of 0.0, 0.1, 0.2, 0.5 and 0.86 wt % C between 25 and 1350°C were determined. An anomalously large mass transport effect observed in two-phase alloy during thermotransport experiments was investigated for the Fe-Ni-C system.

- 2015 (ITB--76.08) DECAY HEAT OF FISSION PRODUCTS AND ACTINIDES OF THE SNR 300. Groenefeld, G. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). Jan 1976. 289p. (In German). Dep. NTIS (US Sales Only), PC A13/MF A01.

The computer code RASPA, which calculates build-up and decay of fission products and actinides is described. The verification of the code and its library has been performed by comparison with theoretical and experimental results of other authors whereby a good agreement has been achieved. Furthermore, an error analysis has shown that the error of the calculated decay heat, which is induced by uncertainties of nuclear data, is less than 10% up to decay times of one month. The time dependence decay heat and the gamma source strength in various zones of the core Mark IA and Mark II of the SNR 300 have been calculated with RASPA. The results are documented and discussed in detail.

- 2016 (IWGFR--3) INTERNATIONAL WORKING GROUP ON FAST REACTORS EIGHTH ANNUAL MEETING. SUMMARY REPORT, PART II. (International Atomic Energy Agency, Vienna (Austria). International Working Group on Fast Reactors). 1975. 114p. (CONF-7504136--P2). Dep. NTIS, PC A06/MF A01.

From 8. meeting international working group on fast reactors; Vienna, Austria (15 Apr 1975).

The various national programs in fast reactor research and development are reviewed.

- 2017 (IWGFR--5) INTERNATIONAL WORKING GROUP ON FAST REACTORS. (International Atomic Energy Agency, Vienna (Austria). International Working Group on Fast Reactors). 1975. 193p. (CONF-750527--(Summ.)). Dep. NTIS (US Sales Only), PC A09/MF A01.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

Separate abstracts were prepared for 18 of the 23 papers included. Abstracts have appeared previously for the remaining 5 papers.

- 2018 (IWGFR--5, pp 18-24) FUEL CLAD FAILURES IN RAPSODIE REACTOR AND AN EXPERIMENTAL IRRADIATION PROGRAM IN ORDER TO OBTAIN VOLUNTARY AND SPONTANEOUS FAILURES. Chalony, A.; Chantoin, P. 1975.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

An experimental program at the RAPSODIE Reactor for the study of FBR fuel pin failure is described. The results to date are also presented.

2019 (IWGFR--5, pp 30-34) STUDY OF FUEL FAILURE DUE TO MANUFACTURING FAULTS.

Gatesope, J.P.; Perret, F. 1975.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

The mechanism for FBR fuel pin failure following sodium entry into the pin through a defective weld is discussed.

2020 (IWGFR--5, pp 35-37) CONTRIBUTION TO THE SPECIALISTS MEETING ON FUEL FAILURE MECHANISMS. Goetzmann, O. 1975.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

The mechanism for inner cladding attack in FBR fuel pins is discussed.

2021 (IWGFR--5, pp 43-49) COMPLEMENTARY INFORMATION TO SUMMARY REPORT ON FUEL FAILURES, THEORETICAL APPROACH, AND SPECIAL PROGRAMS. Mayer, H. 1975.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

The objective of this note is to provide recent complementary information on the status of DFR-455 and Mol-7B and to cover some additional aspects of theoretical treatment of fuel failure mechanisms. Topics of an initiated safety related program are mentioned without detailed discussion.

2022 (IWGFR--5, pp 50-58) SUMMARY ON FUEL PIN FAILURES AND THEORETICAL APPROACH. Mayer, H. 1975.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

It is the purpose of this report to summarize the essential experimental facts and to give some hypothesis for interpretation as far as possible today. The status of theoretical description of fuel pin behaviour (code IAMBUS) is given with emphasis on a partial interpretation of failures. Both of course cannot be conclusive, but will show the main lines of thinking to be followed in the future. The findings and results reported are part of the joint fuel development program for the SNR-300 (Kernkraftwerk Kalkar).

2023 (IWGFR--5, pp 59-64) FUEL FAILURE STUDIES AND EXPERIENCES IN ITALY. Cicognani, G. 1975.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

Of great importance in the design of the fast breeder fuel element are the assumptions and the hypotheses on the statistical behaviour of the fuel pin under both nominal and transient conditions. It is well known that the assumed pin failure limits, as well as the behaviour of a defected pin under operation, affect the average conditions of the core. Therefore a better understanding of the pin failure and damage propagation related mechanisms is mandatory to improve the design criteria and specifications for the fuel and, finally, the economics of the system. A short review the activities carried out in this field by the Fast Reactor Department of the CNEN, both in-pile and out-of-pile, and of the effort on the calculation models and codes recently started by NIRA is presented.

2024 (IWGFR--5, pp 69-82) PNC IRRADIATION

PROGRAMS. Kawata, T.; Kikuchi, S.; Uematsu, K. 1975.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

The Power Reactor and Nuclear Fuel Development Corporation has been conducting various irradiation experiments for fuels and materials in support of design and construction of 'JOYO' experimental fast reactor and 'MONJU' prototype fast reactor.

2025 (IWGFR--5, pp 83-84) PIPER: A NEW FAST REACTOR FUEL PIN PERFORMANCE CODE. Kikuchi, S.; Yokouchi, Y. 1975.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

A new computer program (PIPER=Pin Performance) has been developed for analyzing and predicting LMFBR fuel pin performance with burnup. The objective of the PIPER program is to calculate the fuel pin performance including stresses and deformation in a reactor as the result of a history of power, coolant temperature, coolant flow and other operating conditions. The PIPER program treats the quasi-static transient analysis of a fuel element in the form of a long metallic tube which is the clad for a cylindrical column of porous fuel. Encapsulated pins can be treated as an option. The geometry is axisymmetric and all stresses, strains, forces and deflections are assumed to be axisymmetric. The fuel pin can be divided into as many as ten arbitrary axial sections plus one axial section to treat the fission gas plenum. The fuel region can be divided into up to twenty equal mass radial regions for all calculations except the stress and deformation calculations. To limit the computer running time required for the stress-strain calculations, the fuel region radial nodes are grouped into three regions for the stress-deformation calculations. Mechanical properties, temperature, porosity and concentration dependent models are appropriately weighted for each of these three regions.

2026 (IWGFR--5, pp 152-157) EXPERIMENTAL STUDY OF FAST REACTOR FUEL PIN PERFORMANCE. Davydov, E.F.; Kuzmin, V.I.; Lebedev, I.G.; Syuzev, V.N. 1975.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

Presented in this report are the results of extensive tests of fast fuel pin performance carried out in the BOR-60 reactor. Oxide fuel pins of the actual design were successfully irradiated to 10% burnup of heavy atoms. The 10 to 11% burnup fuel cladding mechanical strain was found to be not in excess of 0.5% and 1.3% for oxide and carbide fuel pins, respectively. The fuel pin characteristics are presented for those with failed cladding. The analysis of failures showed that they were caused mainly by cladding embrittlement resulted from chemical effects of impurities present in fuel. The failures were observed to occur under both steady-state and transient conditions. The further study program is presented aimed at the optimization of the oxide fuel element design and fabrication technology and testing several carbide fuel pin versions.

2027 (IWGFR--5, pp 158-174) BASIC PRINCIPLES OF PROGNOSTICATION FOR OXIDE FUEL ELEMENT WORKING ABILITY OF SODIUM COOLED FAST REACTORS. Golovnin, I.S.; Likhatchev, J.I. 1975.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

The basic parameter influencing the working ability of oxide fuel elements in sodium cooled fast reactors are determined, the principles for the choice of their optimal values are suggested and the main changes appearing in the oxide fuel element during its operation are discussed. The dynamic scale models of thermal-physical and strength estimation of fuel element design are given as well as the criteria for operational reliability. The data is given proceeding from the theoretical conceptions and experimental results on the behaviour of materials under in-off pile conditions.

- 2028 (IWGFR--5, pp 25-29) FUEL FAILURE ON THE RAPSODIE SUBASSEMBLY 'CAPRICORN 1.B'. Chalony, A.; Chantoin, P.; Cauvin, J.C.; Pages, J.P. 1975.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

Fuel pin failure in a 3-pin experimental subassembly in RAPSODIE is described. The location, removal, and post-irradiation examination of the subassembly is described and the postulated failure mechanism is discussed.

- 2029 (IWGFR--5, pp 38-42) SIMULATION OF DIFFERENT TRANSIENT CLAD FAILURE MECHANISMS IN FUEL PIN MODELLING. Kuczera, B. 1975.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

For the analysis of hypothetical core disruptive accidents (HCDAs), failure time and failure position of a fuel pin and the state of the fuel itself are important parameters with respect to the initial conditions of the subsequent fuel coolant interaction (FCI). Various, time dependent load conditions may cause the cladding tube to fail. The main failure reasons are: (1) mechanical fuel-clad interaction due to differential expansion, (2) loading due to fission gas pressure, (3) local contact of molten fuel and clad, and (4) clad melting under dry-out conditions. The influence of threshold values (failure criteria) on simulated accident sequences is demonstrated.

- 2030 (IWGFR--5, pp 67-68) MONJU FUEL PIN SAFETY IRRADIATION PROGRAM IN GETR. Kawata, T.; Kikuchi, S.; Uematsu, K. 1975.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

Within the activities of experimental support of the MONJU safety design, a total of twenty fuel pin tests are planned to be performed in GETR. Of these twenty tests, nine are of the loss of coolant flow (LOC) type, and six are of the local flow blockage (LFB) type. These tests include the experiments on the fresh pins as well as on the fuel pins preirradiated in the fast flux conditions. French fast reactor Rapsodie will be used for the preirradiation of the fuel pins. Remaining five test are of the overpower (OP) type and include only fresh pins. This synopsis gives the outlines of these three types of experiments.

- 2031 (IWGFR--5, pp 142-151) LABORATORY STUDIES ON MELTING AND GAS RELEASE BEHAVIOR OF IRRADIATED FUEL. Weber, E.R.; Slagle, D.D.; Hinman, C.A. 1975.

From International working group on fuel

failure mechanisms; Seattle, WA, USA (11 May 1975).

The response of irradiated fuels to rapid heating is a major factor which must be considered in safety analyses for fast reactors. A series of experiments at HEDL provide observations and data on the response of fast flux (EBR-II) irradiated fuel materials when subjected to rapid heating in the laboratory. Fuel samples within a resistively heated tungsten tube are subjected to thermal transients in a system providing for temperature and pressure measurements, in addition to continuous gamma radiography of the sample. Results of heating fuel at rates of 50° to 250°C/sec. are presented. Unirradiated UO₂ slumps rapidly when melted. Lower power irradiated fuel with uniform fission gas distribution was observed to break apart below the melting point, resulting in dispersal from the effect of released fission gas. Extensive gas bubble formation occurs below the solidus and bubble expansion results in a foam-like structure at the melting temperature. High power irradiated fuel behaves differently, consistent with the relative gas distribution and content. Less fuel motion is observed; the major effect of fission gas is to slow downward displacement upon melting. Measurements of the temperature dependence of gas release indicate that gas release begins at fuel temperatures near 2000°C and a significant fraction of the fission gas has been released when fuel reaches the melting temperature.

- 2032 (IWGFR--5, pp 174-189) IN-REACTOR SAFETY TESTS WITH FUELS FOR FAST NEUTRON REACTORS. Bailly, J.; Kessler, G.; Teague, N.J. 1975.

From International working group on fuel failure mechanisms; Seattle, WA, USA (11 May 1975).

The CEA has realized, alone and in collaboration with various foreign organizations, tests in in-pile loops for supporting the safety analyses of fast reactors. The first part of the SCARABEE program dedicated to cooling accidents was completed in 1974 at CEN-Cadarache. The main results obtained and the efforts in progress to interpret them are described.

- 2033 (IWGFR--8) NINTH ANNUAL MEETING, VIENNA, AUSTRIA, 30 MARCH--2 APRIL 1976. SUMMARY REPORT. PART I. (International Atomic Energy Agency, Vienna (Austria). International Working Group on Fast Reactors). Jun 1976. 26p. (CONF-760396--P1(summ.)). Dep. NTIS (US Sales Only), PC A03/MF A01.

From 9. meeting of the international working group on fast reactors; Vienna, Austria (30 Mar 1976).

This document constitutes the summary report of the Ninth Annual Meeting of the International Working Group on Fast Reactors.

- 2034 (IWGFR--9, pp 87-89) EROSION BY CAVITATION IN THE DIFFUSER OF A MECHANICAL PUMP. Cochet, J.C.; Schaller, K.H. Jun 1976.

From Specialist meeting on cavitation in sodium and studies of analogy with water as compared to sodium; Cadarache, France (12 Apr 1976).

CAVITATION; EROSION; PUMPS; SODIUM; SPECIFICATIONS; LMFBR TYPE REACTORS; PERFORMANCE

- 2035 (IWGFR--9, pp 107) DETECTION AND EVALUATION OF THE DEVELOPMENT OF CAVITATION BY DIRECT LISTENING OF THE ULTRASONIC FIELD WITH

THE HELP OF AN HETERIDYNE FREQUENCY SHIFTING SYSTEM. Danel, F.; Lecoffre, Y. Jun 1976.

From Specialist meeting on cavitation in sodium and studies of analogy with water as compared to sodium; Cadarache, France (12 Apr 1976).

LMFBR. CAVITATION; HETERODYNE RECEIVERS; MATERIALS TESTING; LMFBR TYPE REACTORS; PRIMARY COOLANT CIRCUITS; NOISE; ULTRASONIC TESTING

2036 (IWGFR--9, pp 75-78) CAVITATION PROBLEMS IN SODIUM VALVES. Elie, X. Jun 1976.

From Specialist meeting on cavitation in sodium and studies of analogy with water as compared to sodium; Cadarache, France (12 Apr 1976).

Cavitation poses few problems for sodium valves in spite of the fact that the loops are not pressurized. This is no doubt due to the low flow velocities in the pipes. For auxiliary loop valves no attempt is made to standardize performances with respect to cavitation. For economic reasons cavitation thresholds are approached with large diameter valves.

2037 (IWGFR--9, pp 9-24) CAVITATION PROBLEMS IN MIXING DEVICES OF SNR-300 FUEL ELEMENTS. Benemann, A. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bensberg/Koeln (Germany, F.R.)). Jun 1976.

From Specialist meeting on cavitation in sodium and studies of analogy with water as compared to sodium; Cadarache, France (12 Apr 1976).

Because of a complex flow path within the mixing device developed for the fuel elements of the SNR-300, in order to determine the minimum allowable interval to the beginning of cavitation, experimental tests with the original geometry are necessary. These conclusions show that for cavitation values $CV > 1,3$ in the model and prototype - no cavitation zones can form. For reactor conditions a maximum velocity of $V_{sub(max)} = 4,7 \text{ m/sec}$ is therefore allowable in the free annular space of the compensator unit which corresponds to a massflow of $M = 22,5 \text{ kg/sec}$. A cavitation value of $CV = 1,5$ can be figured for the 120% load factor ($M = 20,4 \text{ kg/sec}$, $T = 560^\circ\text{C}$). The mixing device developed is free of cavitation under the present conditions in the SNR-300.

2038 (IWGFR--9, pp 25-33) CAVITATION PROBLEMS IN FAST REACTOR FUEL SUBASSEMBLIES. Bisci, R.; Muret, G.; Teulon, J. Jun 1976.

From Specialist meeting on cavitation in sodium and studies of analogy with water as compared to sodium; Cadarache, France (12 Apr 1976).

The main reasons are presented why sodium flow in Fast Reactors must be free from cavitation phenomena. Fuel subassembly cavitation problems are described and especially those encountered with flow regulation systems. The choice of a cavitation criterion is discussed: it is assumed that the beginning of cavitation in sodium flow can be extrapolated from measurements on water mock-ups. It is also assumed that the Thoma Number, which characterizes the beginning of cavitation, is strictly independent of the fluid nature.

2039 (IWGFR--9, pp 92-94) CAVITATION WORK CONDUCTED AT RISLEY ENGINEERING AND MATERIALS LABORATORIES. Collinson, A.E. (UKAEA). Jun 1976.

From Specialist meeting on cavitation in sodium and studies of analogy with water as

compared to sodium; Cadarache, France (12 Apr 1976).

BACKGROUND NOISE; CAVITATION; EROSION; PFR REACTOR; PRESSURE DROP; PUMPS

2040 (IWGFR--9, pp 94-101) NERATOOM WORK ON PUMP DEVELOPMENT. Hoornweg, C.J. (Neratoom N.V., The Hague (Netherlands)). Jun 1976.

From Specialist meeting on cavitation in sodium and studies of analogy with water as compared to sodium; Cadarache, France (12 Apr 1976).

The prototype pump has been manufactured by Stork Engineerings Works at Hengelo in 1969. The full-scale test on water has been carried out as part of the procedures of acceptance. Tests on sodium have been carried out in the pump test facility of Interatom at Bensberg (W. Germany); these tests started in March 1971 and were finished in October 1972. During that period nearly 6000 hours of pump testing were accomplished, of which 150 hours the pump was subjected to cavitation. During 30 hours the pump was subjected to a cavitation intensity of more than 3% loss of delivery head.

2041 (IWGFR--9, pp 89-92) EFFECT OF PUMP CAVITATION ON THE DESIGN OF THE PRIMARY PUMPS FOR C.F.R. Worster, R.C. (Weir Pumps Ltd., Alloa (UK)). Jun 1976.

From Specialist meeting on cavitation in sodium and studies of analogy with water as compared to sodium; Cadarache, France (12 Apr 1976).

In the design appraisal of the sodium pumps for the primary circuit of the proposed 1300 MW(e) CFR it has been recognised that cavitation, its effects and its control, is the outstanding hydraulic design problem. Careful consideration of this problem and of the possible effects of pump cavitation on the performance of other reactor systems has led to the conclusion that it is more prudent at present to specify pumps with zero cavitation at normal full speed operating conditions. Under abnormal operation it may be necessary to reduce the pumps' speed to prevent cavitation in the pumps or associated equipment. The principal reasons for this decision were uncertainties concerning the possibility of erosion due to limited cavitation in sodium and the possibility of pump cavitation noise interfering with acoustic detection of malfunctioning of reactor components or of boiling in the reactor core.

2042 (IWGFR--9, pp 62-67) ANALOGY OF WATER AS COMPARED TO SODIUM IN CAVITATION. Bisci, R.; Courbiere, P. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides). Jun 1976.

From Specialist meeting on cavitation in sodium and studies of analogy with water as compared to sodium; Cadarache, France (12 Apr 1976).

After outlining the major aspects of the cavitation research and test program undertaken by the CEA(3) (survey of the parameters defining the onset of cavitation in water and sodium flow, and the consequences of operation under sustained cavitating conditions on sodium reactor structural components), this paper describes the test and measuring equipment that has been developed for such studies. The results of the initial tests in water and in sodium using a thin 20 mm dia. orifices plate, are then presented. Except for uncertainties about the measurements themselves, the cavitation threshold values in cold water and in sodium at temperatures above 400°C has shown

rather good concordance. Testing is currently in progress to confirm these findings. (1) EEC engineer assigned to DRNR/SEDC, (2) CEA engineer (DRNR/STRS), (3) French Atomic Energy Commission.

- 2043 (IWGFR--9, pp 46-57) SURVEY OF LMFBR CAVITATION TECHNOLOGY IN THE U.S.A. Cha, Y.S.; Huebotter, P.R. (Argonne National Lab., Ill. (USA)); Hopenfeld, J. (Energy Research and Development Administration, Washington, D.C. (USA)). Div. of Reactor Research and Development). Jun 1976.

From Specialist meeting on cavitation in sodium and studies of analogy with water as compared to sodium; Cadarache, France (12 Apr 1976).

Several experimental programmes of a basic and applied nature were established in the USA in order to develop guidelines to ensure design and operation of LMFBR hydraulic components free from cavitation and/or cavitation damage. As of March 1976, most of these experimental programs are still in progress. Each programme is briefly described. The available interim data are presented. References that are relevant are provided.

- 2044 (IWGFR--9, pp 33-41) ONSET OF CAVITATION IN PRESSURE DROPPING DEVICES IN WATER AND SODIUM. Collinson, A.E. (UKAEA). Jun 1976.

From Specialist meeting on cavitation in sodium and studies of analogy with water as compared to sodium; Cadarache, France (12 Apr 1976).

It is necessary to develop pressure dropping devices for the control of flow in LMFBR sub-assemblies and other components in and around the core. It is generally required that these devices should have a high flow rate, high pressure loss, occupy a very limited space and not be prone to cavitation. If it is intended to detect coolant boiling acoustically then it is important that there should be no cavitation noise background which would result in reduced sensitivity of the boiling noise monitor. This paper describes a series of tests conducted in water and sodium for the development of cavitation noise-free pressure dropping components for PFR. Initial tests were conducted on single ISA nozzles, being followed by the more practical multi-hole nozzle plates and wire meshes. Tests of a plug in a cylindrical bush are described as an example of a sealing unit with an annular gap.

- 2045 (IWGFR--9, pp 67-75) CAVITATION EXPERIMENTS AT BUTTERFLY VALVES IN WATER AND IN SODIUM AT 850K. Mendte, K.; Klemm, J. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bensberg/Koeln (Germany, F.R.)). Jun 1976.

From Specialist meeting on cavitation in sodium and studies of analogy with water as compared to sodium; Cadarache, France (12 Apr 1976).

Throttling valves fabricated by Gebrueder Adams, Bochum will be installed in the SNR-300 plant, serving as flow control devices during postcram operation. Valves of the same type were used in the pump test facility at INTERATOM (APB) to throttle down the sodium pump head. This test loop was to demonstrate at the same time that the prototype SNR valves could withstand the design conditions (flow and temperature) during endurance tests. To optimize the pump test facility design, a 350mm diameter valve was tested in water under cavitation conditions. With the aid of calculational method described herein, the

hydraulic kinetic relationships of the throttled flow could predict the influences leading to cavitation. The results of the calculational model showed close agreement with that, actually incurred during visual observations. Valves of the same type and in 350mm as well as in 600mm diameter sizes were again cavitation tested in sodium. The cavitation bubbles. b) The primary function of these valves in the test loop APB is to dissipate the pressure head of the pump being tested. Through energy dissipation across the valves, they become strong sources of noise generation, in broad-band frequency levels at these locations up to a factor 50 larger than those at the pump. The cavitation dependent vaporizing noise, developed by incipient cavitation, was, inspite of background noise, measurable. Similar to that for the water test runs, incipient cavitation was detected at the valves when the corresponding sigma worths were between 0.15 to 0.20.

- 2046 (IWGFR--9, pp 57-62) ACTUAL STATUS OF SODIUM CAVITATION STUDIES IN JAPAN. Yamamoto, K. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)); Kamiyama, S.; Hashimoto, H. (Tohoku Univ., Sendai (Japan)). Inst. of High Speed Mechanics); Mochizuki, K.; Nakai, Y.; Ishibashi, E.; Tamaoki, T. Jun 1976.

From Specialist meeting on cavitation in sodium and studies of analogy with water as compared to sodium; Cadarache, France (12 Apr 1976).

A cavitation test has been conducted on some components of the fast experimental reactor JOYO. Design is in progress for the fast prototype reactor MONJU. Deliberate consideration has been taken against cavitation as this reactor will be operated under severer service conditions than that of JOYO. A cavitation test of entrance nozzles of MONJU fuel subassemblies was performed in water. In order to obtain design data a program of cavitation tests is planned.

- 2047 (IWGFR--10, pp 93-97) CLOSED CIRCUIT TELEVISION EQUIPMENT DEVELOPED BY BERKELEY NUCLEAR LABORATORY FOR USE ON THE DOUNREAY PROTOTYPE FAST REACTOR. Friend, D.B.; Jones, A. Jul 1976.

From Meeting on inservice-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

CAMERAS;EVAPORATORS;HEATERS;INSPECTION;LMFBR TYPE REACTORS;PFR REACTOR;REMOTE VIEWING EQUIPMENT;TELEVISION

- 2048 (IWGFR--10, pp 158-159) IN SODIUM MATERIALS MONITORING FOR LMFBR. Blewden, H.J.; Probert, K.S. Jul 1976.

From Meeting on inservice-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

LMFBR TYPE REACTORS;MATERIALS TESTING;IN-SERVICE INSPECTION;REACTOR INTERNALS;SODIUM

- 2049 (IWGFR--10, pp 57-71) ULTRASONIC TESTING OF AUSTENITIC COMPONENTS OF SODIUM COOLED FAST REACTORS. Neumann, E.; Kuhlrow, B.; Roemer, M.; Matthies, K. Jul 1976.

From Meeting on inservice-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

A special ultrasonic technique for testing austenitic steel components of sodium cooled fast reactors under conditions of strong gamma activity and high temperature is described.

2050 (IWGFR--10, pp 9-23) PROPOSAL FOR IN-SERVICE INSPECTION AND MONITORING OF SELECTED COMPONENTS LOCATED WITHIN OR PART OF THE PRIMARY CONTAINMENT OF SODIUM COOLED FAST REACTORS. Bolt, P.R. Jul 1976.

From Meeting on inservice-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

Design and operational experience of CEGB gas cooled reactors and certain overseas reactor plant is reviewed in relation to inservice inspection and monitoring of selected components located within or part of the primary containment of sodium cooled fast reactors. Specific comments are made on the items of further design and development work believed to be necessary.

2051 (IWGFR--10, pp 71-73) ULTRASONIC TESTING, STATE OF THE ART AND POSSIBLE DEVELOPMENTS. Haesen, W.; Tromp, Th.J. Jul 1976.

From Meeting on inservice-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

Ultrasonic testing is considered today as an important method for inservice inspection of the LMFBR type reactor components. A survey is made on what is available to the manufacturers of reactor components.

2052 (IWGFR--10, pp 86-93) TELEVISION/STILL CAMERA WITH COMMON OPTICAL SYSTEM FOR REACTOR INSPECTION. Hughes, G.; McBane, P. Jul 1976.

From Meeting on inservice-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

One of the problems of reactor inspection is to obtain permanent high quality records. Video recordings provide a record of poor quality but known content. Still cameras can be used but the frame content is not predictable. Efforts have been made to combine T.V. viewing to align a still camera but a simple combination does not provide the same frame size. The necessity to preset the still camera controls severely restricts the flexibility of operation. A camera has, therefore, been designed which allows a search operation using the T.V. system. When an anomaly is found the still camera controls can be remotely set, an exact record obtained and the search operation continued without removal from the reactor. An application of this camera in the environment of the blanket gas region above the sodium region in PFR at 150°C is described.

2053 (IWGFR--10, pp 26-35) IN-SERVICE INSPECTION FOR 'MONJU'. Abe, S. Jul 1976.

From Meeting on inservice-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

In-service inspection equipment for Liquid Metal Fast Breeder Reactors will be located in many cases in a limited space with high irradiation and high temperature. Design and research of inservice inspection for MONJU is introduced as follows: Inlet pipe of reactor vessel; Reactor vessel; Heat transfer tubes of IHX; Heat transfer tubes of steam generator. The test result is reported.

2054 (IWGFR--10, pp 43-50) INSPECTION REQUIREMENTS AND CONCEPTS FOR LMFBR HEAT EXCHANGERS AND PUMP. A MANUFACTURERS VIEW. de Boer, G.A.; Tromp, Th.J. Jul 1976.

From Meeting on inservice-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

This paper deals with inspection requirements and concepts for liquid metal fast breeder reactor heat exchangers, steam generators and sodium pumps. Improvement of the on-load surveillance methods are specifically recommended. A chart on the levels of inspection is given.

2055 (IWGFR--10, pp 100-116) IN-SERVICE INSPECTION AND MONITORING OF LMFBR IN FRANCE. Bret, A.; Gallet, B.; Tomachevsky, E. Jul 1976. (In French).

From Meeting on inservice-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

In France, the control and monitoring of the LMFBR type reactor is achieved by: temperature, fluxes and reactivity measurements; sodium leakage detection; ultrasonic testing methods; mechanical deformation and stress analysis.

2056 (IWGFR--10, pp 117-122) DEVELOPMENT OF ACOUSTIC MONITORING TECHNIQUES OF LMFBRs. Burtun, E.J. Jul 1976.

From Meeting on inservice-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

Manufacturing defects may be identified by ultrasonic inspection or by stress wave emission. Mechanical failures arising from maloperation may be detected by visualisation under sodium using ultrasonics. Mechanical failures arising from material loss by corrosion or erosion may be anticipated by the detection of acoustic events. Overheating leading to core damage may be detected from the noise of sodium boiling. Mechanical failure from fatigue may be anticipated from an understanding of the vibrational modes of the sodium and its coupling with the structure.

2057 (IWGFR--10, pp 132-139) AN-APPROACH TO THE ON-LOAD SURVEILLANCE OF SODIUM COOLED REACTORS BY ACOUSTIC METHODS. Foerster, K. Jul 1976.

From Meeting on inservice-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

Among a number of various tasks which are partially more conventional or in a similar way valid for other types of reactors, three topics were thought worth while to be considered for the development of acoustic control systems for fast reactors: 1) sodium boiling detection in reactor core, 2) detection of sodium-water reactions in steam generators and 3) crack detection by stress wave emission analysis.

2058 (IWGFR--10, pp 150-157) RADIOGRAPHIC AND ULTRASONIC INSPECTION OF AN AUSTENITIC PUMP BARREL. Haesen, W.; Klinkert, G.F.; Tromp, Th.J. (Neratoom N.V., The Hague (Netherlands)). Jul 1976.

From Meeting on inservice-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

Volumetric testing of the welds of a radioactive pump barrel poses quite a few problems. This paper describes the series of preliminary tests for the sodium pumps of SNR-300. The tests involved were: Ultrasonic; radiographic.

2059 (IWGFR--10, pp 35-42) IN-SERVICE INSPECTION AT THE SNR 300 WITH THE RANGE JF REACTOR VESSEL. Leder, H. Jul 1976.

From Meeting on inservice-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

A general view of the in-service inspection situation of the SNR 300 within the reactor vessel is given. Specification instructions and recommendations, including the licensing procedure, are shown.

- 2060 (IWGFR--10, pp 79-84) STUDIES IN NON-DESTRUCTIVE TESTING WITH POTENTIAL FOR IN-SERVICE INSPECTION OF LMFBRs. McClung, R.W. Jul 1976.

From Meeting on in-service-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

A variety of nondestructive examination techniques have been and are being developed at ORNL with potential for ISI (In Service Inspection) in LMFBRs. Among these are radiographic techniques for radiation environment and image enhancement, advanced eddy-current techniques and equipment for flaw detection and thickness measurement and ISI of steam generator tubing, and ultrasonic methods for quantitative flaw evaluation using frequency-analysis and bore-side ultrasonic techniques for steam generator tubing. Further developments should result in positive application to ISI.

- 2061 (IWGFR--10, pp 144-146) VIBRATION-ACCELERATION MEASUREMENTS ON PROTOTYPE STEAM GENERATORS. Tromp, Th.J. (Neratom N.V., The Hague (Netherlands)). Jul 1976.

From Meeting on in-service-inspection and monitoring of LMFBRs; Bensberg, German Democratic Republic (9 Mar 1976).

Vibration-acceleration measurements on prototype steam generators of the sodium cooled fast reactor SNR-300 are reported. Vibration measurements were carried out in a sodium component test facility of 50MW. It was shown that no vibration difficulties would arise in the steam generators.

- 2062 (IWGFR--10, pp 139-143) APPLIANCE OF STRESS WAVE EMISSION IN PHENIX REACTOR. Tomachevsky, E.G. Jul 1976. (In French).

Specialists meeting on in-service inspection and monitoring of LMFBRs, Bensberg, Germany, F.R., March 9-11, 1976. Summary report.

The acoustic testing techniques developed in PHENIX are outlined. Feasibility is discussed.

- 2063 (IWGFR--11) SPECIALISTS MEETING ON HIGH TEMPERATURE STRUCTURAL DESIGN TECHNOLOGY OF LMFBRs, CHAMPION, PENNSYLVANIA, U.S.A., APRIL 27-30, 1976. SUMMARY REPORT. (International Atomic Energy Agency, Vienna (Austria). International Working Group on Fast Reactors). Oct 1976. 259p. (CONF-7604147--). Dep. NTIS (US Sales Only), PC A12/MF A01.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

Separate abstracts were prepared for each of the papers included.

- 2064 (IWGFR--11, pp 25-29) INFLUENCE OF CREEP-FATIGUE INTERACTION ON BEHAVIOUR OF TYPE 304 STAINLESS STEEL. Breittling, H.; Grosser, E.D.; Lorenz, H. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

The influence of creep-fatigue interaction on creep and fatigue properties of the SNR-300 structural steel X 6 CrNi 1811 (similar to AISI 304 ss) has been studied. Specimens were either precrept and then fatigue loaded or prefatigued and then creep loaded. In a third type of test

alternating creep fatigue loading was applied. Tests were terminated when rupture occurred. The results were evaluated with respect to the validity of a linear creep-fatigue damage rule as specified in ASME CC 1592. The values determined for the damage factor D show a relatively large scatter and depend on the type of test and the relative amounts of creep and fatigue loading. Alternating loading is most damaging with respect to specimen lifetime. An evaluation of the data shows that the values given in ASME CC 1592 can be regarded as representing a lower bound of creep-fatigue interaction effects.

- 2065 (IWGFR--11, pp 75-80) CEASEMT SYSTEM OF FINITE ELEMENT COMPUTER PROGRAMS - USE FOR INELASTIC ANALYSIS IN LIQUID METAL COOLED REACTOR COMPONENTS. Jeanpierre, F.; Charras, T.; Goldstein, S.; Hoffmann, A.; Livolant, M.; Roche, R. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

The system of finite element computer programs CEASEMT is described. Such a system is used to perform inelastic analysis in liquid metal cooled reactor components. The model for non-elastic strains is explained in some detail.

- 2066 (IWGFR--11, pp 106-111) COMPARISON OF TWO APPROXIMATE ANALYSES OF PIPING SYSTEMS IN A STATE OF CREEP. Johnstone, T.L. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

Two approaches to study the state of creep of piping system in LMFBR type reactors are compared: The Kachanov approach (it is assumed that all stresses in the system will relax in the same manner) and the Spence approach.

- 2067 (IWGFR--11, pp 70-74) INELASTIC PIPING FLEXIBILITY ANALYSIS. A BEAM TYPE METHOD. Roche, R.L.; Hoffmann, A. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

The principle and the simple formulation of a beam type method for inelastic piping flexibility analysis is outlined.

- 2068 (IWGFR--11, pp 169-171) TEST FACILITY FOR LARGE-DIAMETER BELLWS. Schallar, K.H. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

This paper reports on the CADARACHE Test facilities for large diameter bellows for sodium cooled fast breeder reactors.

- 2069 (IWGFR--11, pp 58-64) ANALYSIS FOR OUT OF PLANE BENDING ON CURVED PIPES IN CREEP: DEVELOPMENT OF METHOD AND DESIGN FACTORS. Spence, J.; Boyle, J. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

The aim is to study the out of plane bending of a curved pipe in creep. The current situation in elasticity, and the state of research into the creep of piping components, are briefly reviewed. The ultimate goals of the analysis of curved pipes under combined loading is discussed with the conclusion that, from the point of view of system analysis, it is more expedient to calculate an "energy factor"

rather than the classical "flexibility factor". An approximate energy based analysis is presented which utilises a simple power law of creep and direct optimisation methods on a computer. Results for various material and geometric parameters are presented.

2070 (IWGFR--11, pp 7-24) EXPLORATORY TESTING IN SUPPORT OF CONSTITUTIVE EQUATION DEVELOPMENT FOR INELASTIC ANALYSIS.

Swindeman, R.W.; Klueh, R.L. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

This paper describes some of the significant inelastic behavioral features of type 304 stainless steel and of 2 1/4 Cr-1 Mo steel as observed from uniaxial exploratory mechanical testing. Exploratory tests of reference heats of type 304 stainless steel reveal that the initial yield strength is quite low, the work-hardening capability under both monotonic and cyclic straining is high, and the strain rate effects are almost negligible in the temperature range 400 to 600°C (752 to 1112°F). The influence of low stress creep exposure on the subsequent flow stress is of a hardening nature for the conditions of interest. The creep behaviour can be characterized as exhibiting pronounced creep hardening with minimal time-dependent recovery. Depending upon the specific test conditions, plasticity strain can either suppress or accelerate subsequent creep rates. Exploratory tensile and strain-cycling tests on annealed 2 1/4 Cr-1 Mo steel reveal that this material has a relatively high yield strength, low work-hardening capability, and significant response to different strain rates in the temperature range 300 to 600°C (572 to 1112°F). Exposure to low stress creep conditions can produce a loss in the hardening capability. The creep behaviour can be characterized as exhibiting modest creep hardening under low stress conditions but time-dependent recovery can be significant. As with the austenitic steels, plastic strain can either suppress or accelerate creep rates.

2071 (IWGFR--11, pp 112-121) CONSIDERATIONS FOR WELDMENT DESIGN FOR ELEVATED TEMPERATURE APPLICATIONS. Wei, B.C. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

Because of the scarcity of data and because of the metallurgical and structural complexity of the weldments, current design practice does not generally take detailed account of the behaviour of the weldments, either in terms of criteria limits or in terms of deformation behaviour of weldments into structural analyses. Rather, careful consideration is given to the type and location of welds, to weld materials and to examination of welds; and reliance is placed on the factor of safety used in deriving stress limits from base metal properties to guard against failure of welds. In some cases, such as the K-stress indices factors given in Section III of the ASME code for piping and piping components, determination of these constants are based, in part, on fatigue tests of welded pipe joints. Experience has shown that this approach is adequate at subcreep temperatures where design stress limits are, for the most part, intended to prevent gross inelastic deformation.

2072 (IWGFR--11, pp 29-36) SOME ASPECTS OF THE CEA INVESTIGATIONS ON THE MECHANICAL BEHAVIOUR OF 316 STEELS IN THE MEDIUM TEMPERATURE RANGE. Weisz, M. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

This document deals with the creep behaviour under variable stress of 316 steels. It reports results on the primary circuit welds for PHENIX (stress and strains).

2073 (IWGFR--11, pp 104-106) CONCEPTS ON HIGH TEMPERATURE DESIGN ANALYSIS FOR SNR 300. Bieniussa, K.; Zolli, E. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

The paper briefly describes the evolution, the present situation and the next activities on the design of high temperature components of the DEBENELUX prototype fast breeder reactor SNR-300 with particular regard to the design criteria. Elastic structural analyses are performed for the basic design of the components and are supplied by the manufacturer. In agreement with the Safety Experts simplified and/or detailed inelastic analyses of the critical areas are supplied by the prime contractor of the plant. The elastic computations are evaluated on the basis of a set of design rules derived from ASME Code Case Interpretation 1331-4 but with more conservative limits, and the inelastic ones on the basis of the ASME Code Case Interpretation 1592.

2074 (IWGFR--11, pp 196-207) STATIC AND CYCLIC TESTING OF A PFR SECONDARY CIRCUIT PIPE BEND. Boulton, C.F. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

This report describes tests carried out for the United Kingdom Atomic Energy Authority on specimens fabricated from stainless steel pipes and bends. The specimens were full-size replicas of a particular section of the secondary sodium pipework of the prototype fast reactor (PFR) at Dounreay. The specimens were tested under out-of-plane bending, simulating the loading experienced in the reactor under service conditions. The main object of the tests was to determine the onset of incipient buckling under various conditions of loading and temperature. Fatigue tests were carried out entirely at elevated temperature under a strain spectrum simulating the thermal cycling of the pipework in service. The tests have provided sufficient data to validate the stability and fatigue strength of the pipework under thermal loading.

2075 (IWGFR--11, pp 134-169) SOME ASPECTS OF HIGH TEMPERATURE STRUCTURAL DESIGN ON LMFBR SYSTEM AND COMPONENTS. Cesari, F.; Cecchini, F.; Labanti, M. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

A series of tests has been carried out to study thermal and mechanical properties of LMFBR structural materials and components. Testing facilities are described and their specifications are given. Computer codes for analysis of experimental results are used.

2076 (IWGFR--11, pp 171-196) RECENT EFFORTS OF PNC ON CREEP-FATIGUE, CREEP-BUCKLING AND THERMAL RATCHETTING OF THE LMFBR PRIMARY COOLANT SYSTEM COMPONENTS. Nagata, T.; Imazu, A. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA

(27 Apr 1976).

A series of tests has been carried out to study elastic-plastic-creep behaviors of the model structures of the primary coolant piping components under some monotonic or cyclic loading and to investigate creep-buckling properties of the structures caused by accumulated creep strains under monotonic loading or to investigate their creep fatigue interactions under cyclic loadings and thermal ratcheting properties. Outlines of test rigs that are used for the tests, test procedures, test results and future test programs are presented.

- 2077 (IWGFR--11, pp 245-246) METHODS OF ANALYSIS IN THE STRUCTURAL DESIGN OF LMFBR COMPONENTS - FRENCH PRACTICE. Schaller, K.H. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

The method of analysis in the structural design of LMFBR components performed by the French Atomic Energy Commission at Cadarache are outlined.

- 2078 (IWGFR--11, pp 65-69) INELASTIC THERMAL TRANSIENT ANALYSIS OF TUBESHEETS. Sumikawa, M.; Shibato, E.; Shimizu, T.; Kusumoto, S. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

In this paper the following topics are treated: Models to calculate the metal temperature distribution in tubesheets and analysis of inelastic thermal transients.

- 2079 (IWGFR--11, pp 81-85) SIMPLIFIED METHODS OF ANALYSIS. Townley, C.H.A. Oct 1976.

From Meeting on high temperature structural design technology of LMFBR's; Champion, PA, USA (27 Apr 1976).

A simplified method to predict the stress levels and strain ranges for the design of fast reactor components is presented.

- 2080 (IWGFR--11, pp 37-43) DERIVATION OF TIME DEPENDENT DESIGN-VALUES FOR SNR 300 STRUCTURAL MATERIAL. Lorenz, H.; Breittling, H.; de Heesen, E. Oct 1976.

From IAEA international working group on fast reactors; Champion, PA, USA (27 Apr 1976).

Time-dependent design values were derived from long-term creep rupture data for steel X 6 CrNi 1811 in the unwelded and welded condition. The design values had to satisfy the ASME CC 1592 criteria with respect to creep rupture strength, time to reach 1% strain and transition to tertiary creep as well as the requirement of German regulatory rules to properly account for weld behaviour. For the evaluation and extrapolation 2 proven computer programmes were used. The design data derived under consideration of weld joints show relative good agreement with the values of ASME CC 1592. Consideration of welds leads to lower design values above 550°C and 5×10^3 h with the difference between rolled and weld material becoming larger with increasing time and temperature.

- 2081 (IWGFR--12, pp 26-32) PRESENT STATE OF DEVELOPMENT OF THE FAST BREEDER PROJECT IN THE FEDERAL REPUBLIC OF GERMANY, BELGIUM AND THE NETHERLANDS, AND PROGRESS ACHIEVED DURING 1975 AND EARLY 1976. Daeunert, U.; Kessler, G.

Nov 1976.

From 9. meeting of the international working group on fast reactors; Vienna, Austria (30 Mar 1976).

This report outlines the present state of development of the fast breeder project in the Federal Republic of Germany, Belgium and the Netherlands.

- 2082 (IWGFR--12, pp 13-18) REVIEW OF FAST REACTOR PROGRESS IN JAPAN, MARCH 1976. Tomabechi, K.; Hosaka, A. Nov 1976.

From 9. meeting of the international working group on fast reactors; Vienna, Austria (30 Mar 1976).

This paper describes briefly the fast reactor program in Japan. Specifications on MONJU and JOYO fast reactors are given.

- 2083 (IWGFR--12, pp 33-44) DEVELOPMENT OF FRENCH PROGRAMME ON FAST REACTORS, FROM MARCH 1975 TO MARCH 1976. Vautrety, L. Nov 1976.

From 9. meeting of the international working group on fast reactors; Vienna, Austria (30 Mar 1976).

This paper describes the main features of the French program on fast reactors: Super Phenix, Rapsodie and Phenix.

- 2084 (IWGFR--12, pp 44-87) STATUS OF THE U.S.A. LIQUID METAL FAST BREEDER PROGRAMME, APRIL 1976. Wensch, G.W. Nov 1976.

From 9. meeting of the international working group on fast reactors; Vienna, Austria (30 Mar 1976).

This report is a summary of the major projects and activities of the high priority U.S. liquid metal fast breeder reactor program.

- 2085 (IWGFR--13-1) PROFILES OF FACILITIES USED FOR FRB RESEARCH AND TESTING. SUPPLEMENT 1. (International Atomic Energy Agency, Vienna (Austria). International Working Group on Fast Reactors). Mar 1978. 79p. Dep. NTIS, PC A05/MF A01.

The document contains a concise up-to-date description of French supporting LMFBR facilities which were not included in "Profiles of Facilities Used for FBR Research and Testing" reproduced and distributed by the IAEA in December 1976.

- 2086 (IWGFR--18, pp 58-62) COMMISSION OF THE EUROPEAN COMMUNITIES - ACTIVITIES IN THE FIELD OF FAST REACTORS. Balz, W. Jun 1977.

From 10. IAEA working group on fast reactors; Vienna, Austria (29 Mar 1977).

The Commission of the European Communities is performing activities in the field of fast reactor on two lines: a) activities aiming to prepare the commercialization of fast reactors by coordination and collaboration between national programmes. b) the execution of an own programme in the Joint Research Centre at Ispra (Italy) and Karlsruhe (Federal Republic of Germany) in the field of FBR safety and research on Pu-bearing fuel.

- 2087 (IWGFR--18, pp 50-58) PRESENT STATE OF DEVELOPMENT OF THE FAST BREEDER PROJECT IN THE FEDERAL REPUBLIC OF GERMANY, BELGIUM AND THE NETHERLANDS, AND PROGRESS ACHIEVED DURING 1976 AND EARLY 1977. Daeunert, U.; Kessler, G. Jun 1977.

From 10. IAEA working group on fast reactors; Vienna, Austria (29 Mar 1977).

Development of the LMFBR continues to be a major long-term objective of the nuclear

- programme of the DEBENE+ countries. The following milestones of the German-Belgian-Dutch LMFBR programme are reported: status of work on KNK-II; status of construction work on SNR-300; participation of DEBENE utilities in the SUPERPHENIX project and project work on SNR-2; Franco-German cooperation in the field of LMFBR.
- 2088 (IWGFR--18, pp 1-4) STATUS OF THE USSR FAST REACTOR PROGRAMME AS OF MARCH 1977. Krasnojarov, N.V.; Inyutin, E.I. Jun 1977. (In Russian).
From 10. IAEA working group on fast reactors; Vienna, Austria (29 Mar 1977).
Results of experimental and applied research performance of BR-10, BOR-60 and BN-350, construction of BN-600 reactor and initial design of BN-1600 reactor are reviewed.
- 2089 (IWGFR--18, pp 45-50) REVIEW OF FAST REACTOR ACTIVITIES IN ITALY, APRIL 1977. Pierantoni, F. Jun 1977.
From 10. IAEA working group on fast reactors; Vienna, Austria (29 Mar 1977).
In the frame of the cooperation with France, a short outline of the main results, as well as the status of the research and development work in Italy is given.
- 2090 (IWGFR--18, pp 12-35) REVIEW OF THE UK FAST REACTOR PROGRAMME. MARCH 1977. Smith, R.D. Jun 1977.
From 10. IAEA working group on fast reactors; Vienna, Austria (29 Mar 1977).
This paper reports on the Fast Reactor Programme of United Kingdom. These are the main lines: Dounreay Fast Reactor; Prototype Fast Reactor; Commercial Fast Reactor; engineering development; materials development; chemical engineering/sodium technology; fast reactor fuel; fuel cycle; safety; reactor performance study.
- 2091 (IWGFR--18, pp 8-11) REVIEW OF FAST REACTOR PROGRESS IN JAPAN. Tomabechi, K. Jun 1977.
From 10. IAEA working group on fast reactors; Vienna, Austria (29 Mar 1977).
Only highlights and some topics are reported: Using the Fast Critical Assembly (FCA), experiments to evaluate fission rate in the fuel storage position within the 'MONJU' vessel were performed; detectors of hydrogen in the sodium of the steam generator were developed; experiments to stimulate large sodium water reactions which might take place in a steam generator have been carried out; experiments concerning local fuel failure and tests on various components to be used for 'MONJU' have been continued; concerning safety, experiments of sodium burning were performed.
- 2092 (IWGFR--18, pp 35-45) DEVELOPMENT OF FRENCH PROGRAMME ON FAST REACTORS, FROM MARCH 1976 TO MARCH 1977. Vautrey, L. Jun 1977.
From 10. IAEA working group on fast reactors; Vienna, Austria (29 Mar 1977).
The following milestones of the French LMFBR programme are reported: RAPSODIE (irradiation experiments in order to study the behaviour of the different fuel components of a sodium cooled fast reactor); PHENIX (the burn-up of the first core has reached the nominal value of 50,000MWD/T); SUPER-PHENIX (status of construction).
- 2093 (IWGFR--18, pp 4-8) U.S. LMFBR PROGRAMME AS OF 4-1-77. Yevick, J. Jun 1977.
From 10. IAEA working group on fast reactors; Vienna, Austria (29 Mar 1977).
This summary covers the status of LMFBR projects and base programme activities: Experimental Breeder Reactor-II; the Fast Flux Test Facilities; the Clinch River Breeder Reactor Plant; the High Performance Fuel Laboratory; the Hot Pilot Plant.
- 2094 (IWGFR--18, pp 62-64) ACTIVITIES OF THE OECD-NEA IN THE FIELD OF FAST REACTORS. Royen, J. Jun 1977.
Tenth annual meeting, Vienna, Austria, 29 March - 1 April 1977. Summary report. Part II.
The OECD-NEA is performing the following activities in the field of fast reactors: Held ad hoc meetings of senior experts on safety, development and economics of LMFBR type reactors; publishing a Nuclear Safety Research Index (the index is now expanded to cover fast reactors) and distribution; collect test computer programmes, as well as neutron data.
- 2095 (IWGFR--18) TENTH ANNUAL MEETING. (International Atomic Energy Agency, Vienna (Austria). International Working Group on Fast Reactors). May 1976. 5p. (In Russian). (CONF-770398--). Dep. NTIS (US Sales Only), PC A04/MF A01.
From 10. IAEA working group on fast reactors; Vienna, Austria (29 Mar 1977).
Each paper has been separately processed for the Energy Data Base. (DG)
- 2096 (IWGFR--21, pp 123-132) VIBRATION STUDIES OF BN-350 AND BN-600 STEAM GENERATOR TUBE BUNDLES. Fyodorov, V.G.; Lyashenko, V.V.; Sokolov, A.S.; Lukasevitch, B.I.; Testov, I.N.; Titov, V.F. Dec 1977.
From International Atomic Energy Agency, international working group on fast reactors; Argonne, IL, USA (20 Sep 1977).
The report contains results of vibration studies on BN-350 superheater tube bundles on seven-tube simulators and full-scale steam generators, and of BN-600 vaporizer tube bundles on a multi-tube simulator. The discussion covers main vibrational characteristics of tubes of these steam generators, methods of estimation of full-scale steam generator tubes life derived from results of model vibration tests, and results of repeated in-service checks of BN-350 superheaters vibration. The report deals with vibration strength criteria and limiting norms of tubes vibration. It also provides a description or a method of tube vibration measurements, and steam generator designs.
- 2097 (IWGFR--21, pp 197-210) ANALYSIS OF FLOW INDUCED VIBRATION IN HEAT EXCHANGERS. van Beek, A.E. Dec 1977.
From International Atomic Energy Agency, international working group on fast reactors; Argonne, IL, USA (20 Sep 1977).
A description will be given of three different types of heat exchangers developed by the Dutch Nuclear Industry Group 'Neratom' in cooperation with TNO for the sodium-cooled fast breeder reactor SNR-300 at Kalkar. Moreover, the research related with flow induced vibrations carried out by TNO (Organization for Applied Scientific Research) will be presented. The flow induced forces on the tubes of the straight-tube steam generators were measured at the inlet and outlet section where partial cross-flow occurs. With the

measured flow induced forces the response of a tube was calculated as a function of the tube-to-support bush clearances taking into account the non-linear damping effects from the sodium. The theoretical results showed that for this particular design no tube impact damage is to be expected which was confirmed later by a full scale experiment. Special attention will be devoted to the steam generator with helical-coil tube-bundles, where the sodium flows in a counter cross-flow over the tube-bundle. Extensive measurements of the power spectra of the flow induced forces were carried out since no information could be found in the literature. The vibration analysis will be presented and vibration modes of the entire bundle will be compared with experimentally obtained results. Finally a description of the vibration tests to be carried out on the intermediate heat exchanger (IHx) will be presented.

- 2098 (IWGFR--21, pp 212-223) REVIEW OF ANL BASE TECHNOLOGY STUDIES IN SUPPORT OF THE U.S. LMFBR VIBRATION PROGRAM. Wambsganss, M.W.; Chen, S.S.; Mulcahy, T.M.; Shin, Y.S. Dec 1977.

From International Atomic Energy Agency, international working group on fast reactors; Argonne, IL, USA (20 Sep 1977).

Argonne National Laboratory (ANL) is the center for base technology studies of flow induced vibration for the US LMFBR Program. This paper reviews and summarizes published results, reports on the status of ongoing programs, and discusses future needs as outlined in the US LMFBR Vibrations Program Plan.

- 2099 (IWGFR--21, pp 224-234) INVESTIGATION OF ADDED MASSES AND DAMPING FACTORS FOR VIBRATIONS OF TUBE AND TUBE BUNDLES IN FLUID. Sinyavskii, V.F.; Fedotovskii, V.S.; Kukhtin, A.B. Dec 1977.

From International Atomic Energy Agency, international working group on fast reactors; Argonne, IL, USA (20 Sep 1977).

The vibrations of single cylinders in fluid being surrounded by the solid walls of different form as well as the bundles of cylindric rods have been considered in this report. A model of the hydrodynamic damping of vibrations and the analytic solution of a problem concerning the damping of cylinder vibrations in fluid surrounded by a concentric shell is proposed. The influence of fluid viscosity and vibration frequency on a value of the fluid added mass and the damping factor of vibrations is shown. For the cylinder being vibrated near the boundaries of a special type (for example, a flat wall, a plane gap) and for the cophased vibrations of some triangular and square packed rod bundles there have been carried out the calculations for the added masses and the damping factors by an electrodynamic analogy method. The calculational results are compared with the available experimental data. A satisfactory agreement between the experimental data and the calculational results indicates the possibility for practical using the suggested procedure and the calculational formulae for determining the natural frequencies and the damping factors of vibrations for elastic cylindrical elements in different fluids.

- 2100 (IWGFR--21) SPECIALISTS MEETING ON LMFBR FLOW INDUCED VIBRATIONS. (International Atomic Energy Agency, Vienna (Austria). International Working Group on Fast Reactors). Dec 1977. 237p. (CONF-770933--). Dep.

NTIS, PC A11/MF A01.

From International Atomic Energy Agency, international working group on fast reactors; Argonne, IL, USA (20 Sep 1977).

Separate abstracts were prepared for each of the individual papers included.

- 2101 (IWGFR--21, pp 7-26) FAST REACTOR FLOW INDUCED VIBRATION WITH PARTICULAR REFERENCE TO PFR AND CONCEPTUAL CDFR DESIGN. Collinson, A.E. Dec 1977.

From International Atomic Energy Agency, international working group on fast reactors; Argonne, IL, USA (20 Sep 1977).

This paper describes development work and basic design criteria as applied to vibration induced by fluid flow for the United Kingdom LMFBR programme. Possible harmful effects that could result from excessive vibration levels are highlighted with the associated excitation mechanisms. Instances of flow-induced vibration encountered in development models and in PFR are described. Basic design requirements and the overall philosophy are considered with respect to PFR and notional CDFRs. Ongoing research objectives are outlined giving some insight into the broad strategy involved. Finally, details are given concerning projects presently in progress along with more long term proposals.

- 2102 (IWGFR--21, pp 27-42) FLOW INDUCED VIBRATION STUDIES FOR LMFBR IN JAPAN: PAST AND RECENT STUDIES OF FIV FOR JOYO AND MONJU. Sato, K. Dec 1977.

Specialists meeting on LMFBR flow induced vibrations.

This paper presents the past and recent studies of flow induced vibration of the reactor components for the experimental fast breeder reactor JOYO and the prototype fast breeder reactor MONJU, in which many suggestive results for the higher flow velocity systems in a future reactor are contained. The fuel subassembly is the most important from the view point of the vibration. Thus, the studies were carried out with a mock-up subassembly for JOYO. In this experiment, statistical analysis results of the vibration characteristics of single core subassembly and the effects of external forced vibration, flow disturbance and fuel pin bundle vibration were reported. The further more detailed investigations are now being performed for MONJU. In addition to the above studies, the vibration failure of a sodium valve is reported. The valve is a 8-inch stop valve in sodium flow and heat transfer test loop at O-arai Engineering Center. The failure occurred in 1969 during the performance test of the mechanical pump, and this resulted in a small sodium leak. The cause of the failure was found to be the vibration fatigue of the metal bellows.

- 2103 (IWGFR--21, pp 43-50) VIBRATION MEASUREMENTS AND ANALYSIS FOR THE REACTOR VESSEL AND INTERNALS OF THE SNR-300. Schmidt, H. Dec 1977.

From International Atomic Energy Agency, international working group on fast reactors; Argonne, IL, USA (20 Sep 1977).

During normal operation the SNR-300 LMFBR power plant has a well defined coolant flow distribution with a moderate velocity level. The flow induced vibration potential is therefore very low. Nevertheless much effort has been devoted to study the vibrational behaviour of the reactor internals and the vessel structure. In recent years a number of flow vibration tests in water loops and performance tests in sodium have been conducted

with SNR fuel assemblies. Although some uncertainty will remain due to the reactor environment, the experimental evidence from these tests indicates that flow induced vibration in SNR-300 fuel assemblies will be insignificant. In order to meet the licensing conditions for securing the integrity of the internals, an iterative procedure is planned, consisting of theoretical and experimental structure analysis and a final vibration measurement during reactor commissioning. First results are presented.

- 2104 (IWGFR--21, pp 51-62) SUBSTANTIATION OF VIBRATION STRENGTH OF NUCLEAR REACTOR AND STEAM GENERATOR INTERNALS; MAIN PROBLEMS. Fyodorov, V.G.; Sinyavsky, V.F. Dec 1977.

From International Atomic Energy Agency, international working group on fast reactors; Argonne, IL, USA (20 Sep 1977).

The report details the scope and priority of studies necessary for substantiation of vibration strength of steam generator tube bundles and reactor fuel assemblies, and design modifications helping to reduce flow-induced vibration of the internals specified. Steam generator tube bundles are studied on the basis of a standard establishing vibration requirements at various stages of design, manufacture and operation of a steam generator at a nuclear power station. The main vibration characteristics of tubes obtained through model and full-scale tests are compared with calculation results. Results are provided concerning test-stand vibration tests of fuel elements and fuel assemblies.

- 2105 (IWGFR--21, pp 63-70) FUEL ELEMENT VIBRATIONS IN A FAST REACTOR CORE DUE TO SODIUM FLOW. Teulon, J.; Simonneau, J.P.; Brenir, Y. Dec 1977. (In French).

From International Atomic Energy Agency, international working group on fast reactors; Argonne, IL, USA (20 Sep 1977).

In a fast neutron reactor, the fluid reaches the highest speeds in the core. Very turbulent, the resulting excitation incites the core elements to vibrate. A knowledge of the relative amplitudes and relative lengthening attained is necessary for the evaluation of the damage which could ultimately be produced. These vibration studies are carried out at the CADARACHE Nuclear Center of the totality of the assemblies and on the constituents of these assemblies. Concurrently, an analysis of the fluctuating pressures in the fluid is carried out.

- 2106 (IWGFR--21, pp 99-110) FLOW INDUCED VIBRATION OF THE LARGE-SIZED SODIUM VALVE FOR MONJU. Sato, K. Dec 1977.

From International Atomic Energy Agency, international working group on fast reactors; Argonne, IL, USA (20 Sep 1977).

Measurements have been made on the hydraulic characteristics of the large-sized sodium valves in the hydraulic simulation test loop with water as fluid. The following three prototype sodium valves were tested: (1) 22-inch wedge gate type isolation valve, (2) 22-inch butterfly type isolation valve, and (3) 16-inch butterfly type control valve. In the test, accelerations of flow induced vibrations were measured as a function of flow velocity and disk position. The excitation mechanism of the vibrations is not fully interpreted in these tests due to the complexity of the phenomena, but the experimental results suggest that it closely depends on random pressure fluctuations near the valve disk and flow separation at the contracted cross section between the valve seat

and the disk. The intensity of flow induced vibrations suddenly increases at a certain critical condition, which depends on the type of valve and is proportional to fluid velocity.

- 2107 (IWGFR--21, pp 111-114) HEAT EXCHANGER VIBRATION. Richards, D.J.W. Dec 1977.

From International Atomic Energy Agency, international working group on fast reactors; Argonne, IL, USA (20 Sep 1977).

The heat exchangers of various types are common items of plant in the generation and transmission of electricity. The amount of attention given to the flow-induced vibrations of heat exchangers by designers is usually related to the operational history of similar items of plant. Consequently, if a particular design procedure yields items of plant which behave in a satisfactory manner during their operational life, there is little incentive to improve or refine the design procedure. On the other hand, failures of heat exchangers clearly indicate deficiencies in the design procedures or in the data available to the designer. When such failures are attributable to flow-induced vibrations, the identification of the mechanisms involved is a prime importance. Ideally, basic research work provides the background understanding and the techniques necessary to be able to identify the important mechanisms. In practice, the investigation of a flow-induced vibration problem may identify the presence of mechanisms but may not be able to quantify their effects adequately. In these circumstances the need for additional research work is established and the objectives of the research programme emerge. The purpose of this paper is to outline the background to the current research programme at C.E.R.L. on heat exchanger vibration.

- 2108 (IWGFR--21, pp 115-122) FLOW INDUCED VIBRATION OF SECONDARY PIPING OF LMFBR. Gibert, R.J.; Axisa, F. Dec 1977.

From International Atomic Energy Agency, international working group on fast reactors; Argonne, IL, USA (20 Sep 1977).

This paper presents a method for evaluating the characteristics of vibrations caused by internal flow in three-dimensional piping systems conveying high density fluids. The excitation of the circuit is mainly caused by the flow singularities, and it is shown that the problem may be reduced to calculate the response of the circuit to an acoustical pressure discontinuity, localised at each flow singularity. The theoretical formulation of the coupled acoustical-mechanical problem and to its numerical solution by the computer code TEDEL is discussed. An experimental test of the method is described. The tested piping system consists of a stainless steel tube circuit comprising four 90° bends, conveying water. Vibrations are excited by a half closed gate valve. Satisfactory results are obtained concerning both the frequencies of resonance of the circuit and the level of the vibrations observed.

- 2109 (IWGFR--21, pp 133-164) FFTF SCALE-MODEL CHARACTERIZATION OF FLOW INDUCED VIBRATIONAL RESPONSE OF REACTOR INTERNALS. Ryan, J.A.; Julyk, L.J. Dec 1977.

From International Atomic Energy Agency, international working group on fast reactors; Argonne, IL, USA (20 Sep 1977).

As an integral part of the Fast Test Reactor Vibration Program for Reactor Internals, the flow-induced vibrational characteristics of scaled Fast Test Reactor core internal and peripheral components were assessed under

scaled and simulated prototype flow conditions in the Hydraulic Core Mockup. The Hydraulic Core Mockup, a 0.285 geometric scale model, was designed to model the vibrational and hydraulic characteristics of the Fast Test Reactor. Model component vibrational characteristics were measured and determined over a range of 36% to 111% of the scaled prototype design flow. Selected model and prototype components were shaker tested to establish modal characteristics. The dynamic response of the Hydraulic Core Mockup components exhibited no anomalous flow-rate dependent or modal characteristics, and prototype response predictions were adjudged acceptable.

- 2110 (IWGFR--22) PROPERTIES OF PRIMARY CIRCUIT STRUCTURAL MATERIALS INCLUDING ENVIRONMENTAL EFFECTS. (International Atomic Energy Agency, Vienna (Austria). International Working Group on Fast Reactors). Dec 1977. 238p. (CONF-771052--). Dep. NTIS (US Sales Only), PC A11/MF A01.

From International working group on fast reactors (IAEA/IWGFR) specialists meeting; Bensburg, F.R. Germany (17 Oct 1977).

Summaries are presented of thirty-one papers dealing with structural materials behavior in LMFBR primary coolant circuits. (DG)

- 2111 (IWGFR--23) SODIUM REMOVAL AND DECONTAMINATION. (International Atomic Energy Agency, Vienna (Austria). International Working Group on Fast Reactors). Aug 1978. 214p. (CONF-780232--). Dep. NTIS (US Sales Only), PC A10/MF A01.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

Separate abstracts are included for each of the papers presented concerning sodium removal and decontamination.

- 2112 (IWGFR--23, pp 1-7) STATUS OF SODIUM REMOVAL AND COMPONENT DECONTAMINATION TECHNOLOGY IN THE SNR PROGRAM. Haubold, W. (INTERATOM, Bergisch Gladbach, Germany); Smit, C.C.; Stade, K.C. Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

The experience with sodium removal and component decontamination processes within the framework of the SNR project is discussed. The moist nitrogen process has been successfully applied to remove sodium from all 66 fuel elements of the KNK I core. Progress has been obtained in removing sodium from fuel elements and large components by vacuum distillation. Areas where future development is required are identified.

- 2113 (IWGFR--23, pp 8-12) TNO EXPERIENCE ON SODIUM CLEANING OF LARGE PLANT COMPONENTS BY VACUUM DISTILLATION. Smit, C.C. (Sodium Component Test Facility, Hengelo, Netherlands). Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

The Intermediate Heat Exchanger and Steam generators developed within the framework of the SNR program are being tested in the 50 MW Test facility at Hengelo, The Netherlands. The facility was designed and built by Neratoom, and is operated by TNO, the Dutch Organization for Applied Scientific Research. Sodium technology work is done in close cooperation with Neratoom and with TNO-laboratories at Apeldoorn, where several smaller sodium rigs

and other facilities are available. Some work has been done to develop reliable vacuum distillation procedures for large components, based on the cleaning experience with small plant components. The results of these procedures applied to large plant components are reported in this paper.

- 2114 (IWGFR--23, pp 13-17) CORROSION AND CLEANING ASPECTS OF SODIUM SIDE CREVICES IN COMPONENTS OF LMFBR'S. Chirer, E.G. (Metals Research Inst. TNO, Apeldoorn, Netherlands). Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

The aims of the investigations are: determination of the corrosive aspects of high concentration of caustic reaction products of sodium in crevices on the structural materials of the component; and comparison of the effectiveness of different cleaning procedures with respect to removal of sodium from crevices, e.g., water, steam, alcohol cleaning, vacuum distillation. Materials investigated are the Cr--Mo steels 2-1/4 Cr 1 Mo stabilized with Nb, 9 Cr 1 Mo, 12 Cr 1 Mo and the austenitic stainless steel AISI 304. The experiments are performed either on tubular specimens, consisting of two concentric tubes with crevices varying from a press fit to a positive clearance of approx. 0.05 mm, or on prestressed flat specimen. To observe the behavior of welded joints the specimens are provided with a weld seam.

- 2115 (IWGFR--23, pp 30-37) EXPERIENCE WITH CLEANING OF SODIUM-WETTED COMPONENTS AND DECONTAMINATION AT NUCLEAR RESEARCH CENTRE KARLSRUHE. Menzenhauer, P.; Borgstedt, U.; Stamm, H.H.; Dippel, T.; Kunze, S.; Hentschel, D. (Kernforschungszentrum, Karlsruhe, Germany). Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

Within the framework of the Fast Breeder project various institutes of the Karlsruhe Nuclear Research Center operate sodium loops for corrosion studies and component tests under sodium. The operation of these loops has led to extensive experience in cleaning sodium-wetted components. This experience related to the alcohol method, the removal of sodium by melting, storage in air, and to cleaning by means of steam. Deposition samples from radioactive sodium loops were used for decontamination experiments employing various decontaminating agents. The department concerned with the treatment of radioactive waste studied the use of molten salts and paste type cleansers for components unsuitable for mechanical decontamination, primarily with the objective to reduce the amount of radioactive waste.

- 2116 (IWGFR--23, pp 38-42) SODIUM REMOVAL OF FUEL ELEMENTS BY VACUUM DISTILLATION. Buescher, E.; Haubold, W.; Jansing, W.; Kirchner, G. (INTERATOM, Bergisch Gladbach, Germany). Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

Cleaning of sodium-wetted core components can be performed by using either lead, moist nitrogen, or alcohol. The advantages of these methods for cleaning fuel element without causing damage are well known. The disadvantage is that large amounts of radioactive liquids are formed during handling in the latter two

cases. A new method to clean components is described. The main idea is to remove all liquid metal from the core components within a comparatively short period of time. Fuel elements removed from the reactor must be cooled because of high decay heat release. To date, vacuum distillation of fuel elements has not yet been applied.

- 2117 (IWGFR--23, pp 42-46) INTERATOM EXPERIENCE OF CLEANING SODIUM-WETTED COMPONENTS. Haubold, W. (INTERATOM, Bergisch Gladbach, Germany). Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

INTERATOM has been concerned since 1967 with the development, testing, and application of methods to clean sodium-wetted components by moist nitrogen, vacuum distillation or alcohol. The three cleaning methods mentioned above are practised at present, too, with minor modifications, by INTERATOM with methods of sodium removal since 1973 are summarized.

- 2118 (IWGFR--23, pp 47-50) CONTRIBUTION TO THE CHARACTERIZATION OF RADIOACTIVE SURFACES AFTER SODIUM CORROSION. Menken, G.; Holl, M. (INTERATOM, Bergisch Gladbach, Germany). Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

Since 1972 INTERATOM is performing sodium mass and activity transfer investigations in a SNR-corrosion mockup loop which allows to study the transport of activated corrosion products in the primary heat transfer system of a sodium cooled reactor. The loop simulates the temperature and flow conditions and the materials combination of the SNR 300. The mass transfer examinations were aimed at the determination of the linear corrosion and deposition rates; the selective corrosion of the alloying elements; and the transfer of activated corrosion products.

- 2119 (IWGFR--23, pp 50-53) SHORT REVIEW OF SODIUM REMOVAL EXPERIENCE IN PNC. Kamei, M.; Mukaibo, R. (Osarai Engineering Centre, Tokyo, Japan). Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

Since the establishment of the Power Reactor and Nuclear Fuel Development Corporation (PNC), various research work and development have been carried out for the development of sodium cooled fast breeder reactors. Most of the works, especially those related to large scale facilities are done at the Osarai Engineering Center (OEC) of the PNC. Sodium removal from components or assemblies has been an area with much concern from the beginning of these works. Among the conventional methods, removal by steam, water, alcohol and vacuumization at high temperatures have been done. In Joyo, the experimental fast breeder reactor which reached initial criticality on April, 1977, sodium removal from the grapples of the fuel handling facility and the fuel assemblies which come out of the reactor are done as routine work during refueling.

- 2120 (IWGFR--23, pp 100-107) PARAMETER SURVEY ON THE SATURATION BOILING PROCESS FOR CREVICE SODIUM REMOVAL. Kamei, M.; Kanbe, M.; Nakamura, K.; Fukada, T.; Ishii, Y.; Saito, R.; Mimoto, Y. (Osarai Engineering Centre, Tokyo, Japan). Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

It was found out from the experiences that sodium removal from small crevices by conventional method was difficult. And, in order to improve the operation, a saturation boiling process was applied and experiments using test pieces have started. Since the purpose of this process is to remove final remaining deposits, bulk sodium removal must be done before. The saturation boiling process is a modification of the water charging process. When water is charged into the process cell or tank, the pressure inside is adjusted so that boiling of the water is initiated. The formation of hydro-oxide membranes at the surface of the sodium prevents further reaction between sodium and water. Results have been obtained from test pieces of bolts and crevices of flat plates and cylinders.

- 2121 (IWGFR--23, pp 108-123) UK FAST REACTOR COMPONENTS: SODIUM REMOVAL DECONTAMINATION AND REQUALIFICATION. Donaldson, D.M. (UKAEA, Risley, England); Bray, J.A.; Newson, I.H. Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

Over the past two decades extensive experience on sodium removal techniques has been gained at the UKAEA's Dounreay Nuclear Establishment from both the Dounreay Fast Reactor (DFR) and the Prototype Fast Reactor (PFR). This experience has created confidence that complex components can be cleaned of sodium, maintenance or repair operations carried out, and the components successfully re-used. This past and continuing experience is being used in forming the basis of the plant to be provided for sodium removal, decontamination and requalification of components in the UK's future commercial fast reactors.

- 2122 (IWGFR--23, pp 124-138) PROGRESS IN DEVELOPMENT AND APPLICATION OF METHODS OF WASHING AND DECONTAMINATION OF OBJECTS IN CONTACT WITH SODIUM. Msika, D.; Lafon, A. (Centre de Cadarache, Saint-Paul-lez-Durance, France). Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

In the technology of liquid sodium cooled fast reactors, the necessary processes for washing and decontamination have been demonstrated. For sodium removal, different solutions have been considered and tested in France. The study have been progressively oriented toward defining a process using a fine dispersion of water in a gas (atomization). The results obtained by that method on non-radioactive components were satisfactory insofar as the efficiency and safety of the operation was concerned. The purpose of decontaminating components from the reactor primary circuits is to reduce the level of surface activity to a level compatible with personnel access without biological shielding.

- 2123 (IWGFR--23, pp 139-145) CLEANING OF SMALL COMPONENTS OF COMPLEX GEOMETRY BY MEANS OF THE SODIUM-ALCOHOL REACTION. de Luca, B.; Grasso, C.; Spadoni, M. (CSN, Rome, Italy). Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

The results of some experiments on the vacuum reaction between butylcellosolve and

sodium, contained in small diameter capillaries, are reported. The effects on the cleaning rate of the temperature, amount of solvent, diameter and position of the capillaries are analyzed. The facility, used for the cleaning of small components of complex geometry, is described.

- 2124 (IWGFR--23, pp 145-151) SODIUM COMPONENTS CLEANING STATUS IN THE ITALIAN FAST REACTOR PROGRAM. de Luca, B. (CSN, Rome, Italy); Labanti, V.; Mennucci, M. Aug 1978. From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

As a consequence of the Italian Fast Reactor Development, mainly aimed to the PEC project and to the participation in the French Superphenix project, it is of increasing importance to set up a reliable method of sodium removal for specific reactor components and related test loops. So far the problem has been faced only in few occasions and in any case without contamination.

- 2125 (IWGFR--23, pp 151-153) SUMMARY OF SODIUM REMOVAL AND DECONTAMINATION PROGRAMS IN THE USA. Steele, D.P. (Rockwell International, Canoga Park, CA); Brehm, W.F. Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

The goals of the United States Department of Energy sodium removal and decontamination programs are: (1) to identify plant requirements and develop safe effective processes for sodium removal, both for experimental facilities and for reactor components; (2) to develop effective decontamination processes for removing deposited radioactivity; (3) to establish and put into use a set of workable criteria for equalification and return to service of components after sodium removal and decontamination; (4) to design, build and operate facilities in which to perform the sodium removal and decontamination operations. A summary of progress toward each of these goals is presented.

- 2126 (IWGFR--23, pp 157-164) CHARACTERIZATION OF CORROSION PRODUCT DEPOSITS IN SODIUM SYSTEMS. Colburn, R.P. (Hanford Engineering Development Lab., Richland, WA). Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

The characterization of mass transfer deposits on the containment surface of sodium systems is an essential step in the development of LMFBR decontamination processes. The surfaces of a sodium test facility used to study radioactive mass transport were examined to determine the chemical and physical nature of the deposits. The examination showed two fundamentally different problems will occur in the hot and cold leg regions of an LMFBR heat transport system. In the hot leg ^{60}Co , ^{59}Co and ^{54}Mn will diffuse into the base metal and decontamination may require the removal of several microns of the base metal. In the cold leg ^{54}Mn will be the dominant radionuclide.

- 2127 (IWGFR--23, pp 83-99) EXPERIENCE ON SODIUM REMOVAL FROM VARIOUS COMPONENTS. Kamei, M.; Kanbe, M.; Yagisawa, H.; Sasaki, S.; Kataoka, H.; Fukada, T.; Ishii, Y.; Saito, R.; Mimoto, Y. (Oarai Engineering Centre, Tokyo, Japan). Aug 1978.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

Since 1970, OEC (Oarai Engineering Center) has been investigating the following methods for removal of sodium from the components of sodium plants: steam cleaning for the 50 MW steam generator, secondary proto-type pump of JOYO and dummy fuel assembly of JOYO, alcohol cleaning for sector model of intermediate heat exchanger (IHX) of JOYO and a proto-type isolation valve of JOYO and cleaning by vacuumization at high temperature for Regenerative Heat Exchangers. The outline of the sodium Disposal Facility and experience of sodium removal processing on the 50 MW steam generator, the crevices of the experimental subassemblies, the fuel handling machine of MONJU and the regenerative heat exchanger of the Sodium Flow Test Facility are outlined.

- 2128 (IWGFR--23, pp 164-173) DEVELOPMENT OF ACIDIC PROCESSES FOR DECONTAMINATING LMFBR COMPONENTS. Hill, E.F. (Rockwell International, Canoga Park, CA); Colburn, R.P.; Lutton, J.M.; Maffei, H.P. 1977.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

The objective of the DOE decontamination program is to develop a well-characterized chemical decontamination process for application to LMFBR primary system components that subsequently permits contact maintenance and allows requalification of the components for reuse in reactors. The subtasks of deposit characterization, development of requalification and process acceptance criteria, development of process evaluation techniques and studies which led to a new acidic process for decontaminating 304 stainless steel hot leg components are described.

- 2129 (IWGFR--23, pp 173-181) NON-AQUEOUS REMOVAL OF SODIUM FROM REACTOR COMPONENTS. Welch, F.H.; Steele, D.P. (Rockwell International, Canoga Park, CA). 1977.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

Reactor components from sodium-cooled systems, whether radioactive or not, must have the sodium removed before they can be safely handled for (1) disposal, (2) examination and test, or (3) decontamination, repair, and requalification. In the latter two cases, the sodium must be removed in a manner which will not harm the component, and prevent future use. Two methods for sodium removal using non-aqueous techniques have been studied extensively in the USA in the past few years: the alcohol process, which uses a fully denatured ethanol to react away the sodium; and the evaporative process, which uses heat and vacuum to evaporate the sodium from the component.

- 2130 (IWGFR--23, pp 195-203) EXPERIENCE OF CLEANING AND DECONTAMINATION OF THE BN-350 REACTOR COMPONENTS. Vasilenko, K.T.; Kochetkov, L.A.; Arkhipov, V.M.; Baklushin, R.P.; Gorlov, A.I.; Kiselev, G.V.; Rezinkin, P.S.; Samarkin, A.A.; Tverdokski, N.D. 1977.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

In the course of start-up, adjustment and operation of the BN-350 reactor there arose a need for cleaning from sodium and decontamination of primary and secondary

equipment components. Design schemes of the systems provided for this purpose as well as those specially designed for cleaning of steam generator evaporators are considered. Technological processes of cleaning and decontamination for some reactor components (removable parts of circulating pumps, evaporators, valves) are described, the results are presented.

- 2131 (IWGFR--23, pp 182-195) EBR-2 EXPERIENCE WITH SODIUM CLEANING AND RADIOACTIVITY DECONTAMINATION. Ruth, W.E.; Smith, C.R.F. (Argonne National Lab., IL). 1977.

From IAEA meeting on sodium removal and decontamination; Richland, WA, USA (14 Feb 1978).

The EBR-II is now in its 13th year of operation. During that period more than 2400 subassemblies have been cleaned of sodium without a serious incident of any kind by a two-step process developed at Argonne. Sodium cleaning and decontamination of other reactor components has been performed only on the relatively few occasions in which a repair or replacement has been required. A summary of the EBR-II experience is presented. A new facility is described for the improved cleaning and maintenance of sodium-wetted primary components.

- 2132 (IWGFR--24-1) INTERNATIONAL WORKING GROUP ON FAST REACTORS ELEVENTH ANNUAL MEETING. SUMMARY REPORT, PART I. (International Atomic Energy Agency, Vienna (Austria). International Working Group on Fast Reactors). Jun 1978. 32p. (CONF-780487--P1(Summ.)). Dep. NTIS (US Sales Only), PC A03/MF A01.

From 11. meeting on fast reactors; Bologna, Italy (17 Apr 1978).

The Summary Report (Part I) contains the minutes of the meeting. (DG)

- 2133 (IWGFR--24-3) INTERNATIONAL WORKING GROUP ON FAST REACTORS ELEVENTH ANNUAL MEETING. SUMMARY REPORT, PART III. (International Atomic Energy Agency, Vienna (Austria). International Working Group on Fast Reactors). Sep 1978. 30p. (CONF-780487--P3(Summ.)). Dep. NTIS (US Sales Only), PC A03/MF A01.

From 11. meeting on fast reactors; Bologna, Italy (17 Apr 1978).

The Eleventh Annual Meeting of the IAEA International Working Group on Fast Reactors was held at the Palazzo dei Congressi, Bologna, Italy from 17--20 April 1978. The Summary Report (Part I) contains the Minutes of the Meeting. The Summary Report (Part II) contains the papers which review the national program in the field of LMFBRs and other presentations at the Meeting. The Summary Report (Part III) contains the discussions on the review of the national programs.

- 2134 (IWGFR--26) SPECIALISTS' MEETING ON EQUATIONS OF STATE OF MATERIALS OF RELEVANCE TO THE ANALYSIS OF HYPOTHETICAL FAST BREEDER REACTOR ACCIDENTS. (International Atomic Energy Agency, Vienna (Austria). International Working Group on Fast Reactors). 1978. 143p. (CONF-780614--(Summ.)). Dep. NTIS (US Sales Only), PC A07/MF A01.

From Specialists' meeting on equations of state of materials of relevance to the analysis of hypothetical fast breeder reactor accidents; Harwell, UK (19 Jun 1978).

Summaries are included of ten papers.

- 2135 (IWGFR--27) COMPILATION OF PIPING BENCHMARK PROBLEMS - COOPERATIVE INTERNATIONAL EFFORT. McAfee, W.J. (comp.). (Oak Ridge National Lab., TN (USA)). Jun 1979. 590p. Dep. NTIS (US Sales Only), PC A25/MF A01.

This report is the culmination of an effort initiated in 1976 by the IWGFR to evaluate detailed and simplified analysis methods for piping systems with particular emphasis on piping bends. The procedure was to collect from participating member IWGFR countries descriptions of tests and test results for piping systems or bends, to compile, evaluate, and issue a selected number of these problems for analysis, and to compile and make a preliminary evaluation of the analysis results. A variety of both detailed and simplified analysis solutions were obtained. A brief comparative assessment of the analyses is contained in this report. All solutions submitted have also been included in order to provide users of this report the information necessary to make their own comparisons or evaluations.

- 2136 (JAERI-M--6851) STRATEGIES IN DEVELOPMENT OF ADVANCED FUELS FOR LMFBR. Hada, M. (Japan Atomic Energy Research Inst., Tokyo). Dec 1976. 59p. (In Japanese). Dep. NTIS (US Sales Only), PC A04/MF A01.

Strategies in development of advanced fuels for LMFBR are reviewed. Recent irradiation experiment and out-of-pile test data of the fuels are given in detail. The present status of development of oxide fueled LMFBR is also treated.

- 2137 (JAERI-M--6993) CALCULATION PROGRAM FOR FAST REACTOR DESIGN. 2. MULTI-DIMENSIONAL PERTURBATION THEORY CODE BASED ON DIFFUSION APPROXIMATION: PERKY. Iijima, S.; Yoshida, H.; Sakuragi, H. (Japan Atomic Energy Research Inst., Tokyo). Feb 1977. 51p. (In Japanese). Dep. NTIS (US Sales Only), PC A04/MF A01.

The program PERKY is used for calculating reactivity worth and kinetic parameters of a fast breeder reactor by ordinary first-order or exact perturbation theory based on two- or three-dimensional diffusion approximation. It calculates effective delayed neutron fraction, prompt neutron lifetime and nuclide reactivity worth mapping using microscopic cross-sections, and also reactivity change using microscopic or macroscopic cross-sections. The anisotropic effect of leakage on the reactivity change calculated by using macroscopic cross-sections can be estimated with bidirectional, diffusion coefficients calculated from the Bessist formulation. The real and adjoint neutron fluxes for perturbation calculations may be generated by multi-dimensional diffusion code CITATION, and the microscopic and macroscopic cross-sections be provided by effective cross-section generating code PIGEON.

- 2138 (JAERI-M--7139) ADEQUACY OF THE ANALYSIS OF MOCK-UP CONTROL ROD EXPERIMENT WITH FCA. Mizoo, N.; Nakano, M. (Japan Atomic Energy Research Inst., Tokyo). Jul 1977. 33p. (In Japanese). Dep. NTIS (US Sales Only), PC A03/MF A01.

A method of numerical analysis has been investigated for the mock-up control rod experiment of FCA VII-1 assembly constructed as the engineering mock-up of prototype fast breeder reactor MONJU. The results of criticality and B₀C mock-up control rod worths analysis for the assembly are described in comparison with the experimental ones. The

tendency of the C/E value with ^{10}B enrichment and the interaction effect of the multiple rods array was also examined. Reactivities and the mock-up rods worths were obtained with the X-Y geometry six groups diffusion theory. Twelve kinds of the mock-up rods with different ^{10}B contents and/or enrichments were used in the experiment; effective cross-sections are provided for each rod by calculation using the collision probability method. Criticality of VII-1 90Z assembly is underestimated for 3 reference critical configurations, ranging from $-0.65\%\Delta k/k$ to $-0.77\%\Delta k/k$. The C/E values at core center for 12 kinds of B₄C mock-up rods range from 1.03 to 1.09. The overestimate of the rod worth increases with macroscopic absorption cross-section of the rod region. The C/E values for 24 different arrays of the mock-up rods ranging from single rod to five rods lie between 1.04 and 1.08. The C/E value tends to decrease with increase in the number of rods inserted, the values for five rods arrays being about 4% lower than those for single rod arrays. The calculated interaction effects of the multiple rods arrays are slightly more negative than the experimental ones.

- 2139 (JAERI-M-7280) EXCURS: A COMPUTING PROGRAMME FOR ANALYSIS OF CORE TRANSIENT BEHAVIOUR IN A SODIUM COOLED FAST REACTOR. Saito, S. (Japan Atomic Energy Research Inst., Tokyo). Sep 1977. 67p. Dep. NTIS (US Sales Only), PC A04/MF A01.

In the code EXCURS developed for core transient behaviour calculation of a sodium-cooled fast reactor, a one-channel model is used to represent thermal behaviour of the reactor core. Calculations are made for three different channels; i.e. average, hot and hottest. In the average channel the power density and coolant velocity are equal to the mean values of the whole core. In the hot channel, a maximum power density of the core and a specific coolant velocity are introduced. In the hottest channel, engineering hot channel factors are considered to the hot channel. A one-point neutron kinetics equation with six delayed neutron groups is used to calculate the time-dependent power behaviour. Externally introduced reactivity effect and control rod movement in the case of a scram are taken into account. In the feedback effects evaluated on the basis of the average channel temperatures are considered Doppler effect, fuel axial expansion, cladding expansion, coolant expansion and structure expansion. The decay heat after reactor scram is also considered. Heat balance is taken in each cross section, neglecting the axial heat transfer except for the coolant region. Temperature dependence of the physical properties of materials is considered by second-order polynomials approximation, and also the fuel melting process. Each channel can be divided into a maximum of 20 regions in both radially and axially. The reactor core transient behaviour due to reactivity insertion or loss-of-coolant flow can be studied by EXCURS. The calculated results are plotted optionally by connected code EXPLOT.

- 2140 (JAERI-M-7291) REACTOR ENGINEERING DIVISION ANNUAL REPORT. (APRIL 1, 1976 - MARCH 31, 1977). Hirota, J.; Asaka, T.; Suzuki, T.; Mitani, H.; Akino, F. (eds.). (Japan Atomic Energy Research Inst., Tokyo). Sep 1977. 193p. Dep. NTIS (US Sales Only), PC A10/MF A01.

Research activities in the Division of Reactor Engineering in fiscal 1976 are described. Works of the division concern mainly the development of multi-purpose Very High

Temperature Gas Cooled Reactor, fusion reactor engineering, and the development of Liquid Metal Fast Breeder Reactor in Power Reactor and Nuclear Fuel Development Corporation. Contents of the report are nuclear data and group constants, theoretical method and code development, integral experiment and analysis, shielding, heat transfer and fluid dynamics, reactor and nuclear instrumentation, dynamics analysis and control method development, fusion reactor technology, and activities of the Committee on Reactor Physics.

- 2141 (JAERI-M-7763) INVESTIGATION OF LMFBRs WITH TH BLANKET FOR SUPPLY OF U-233 TO HTGR. Yoshida, H.; Ohta, F. (Japan Atomic Energy Research Inst., Tokyo). Jul 1978. 60p. (In Japanese). Dep. NTIS (US Sales Only), PC A04/MF A01.

A symbiotic energy system between fast breeder reactor and high temperature gas-cooled thermal reactor appears to be promising as an energy system self-sufficient in fuel to produce both electricity and high temperature process heat. In the system, LMFBR supplies sufficient quantity of Pu to keep itself going and also produces U-233 for Th- ^{233}U HTGR. In the investigation to find out such LMFBR with Th blanket having good Pu/U-233 breeding to keep the system in operation through neutronic and fuel cycle analyses, three types of 1000Mwe LMFBR were examined: conventional two-zoned core LMFBR, axial partial heterogeneous core LMFBR, and radial partial heterogeneous core LMFBR. In each type, the UO_2 -blanket is partly replaced by ThO_2 -blanket to produce U-233. Results indicate the axial partial heterogeneous core LMFBR is appropriate for the symbiotic energy system in breeding performance, safety performance (sodium void coefficient etc.) and economy.

- 2142 (JAERI-M-7844) REACTOR ENGINEERING DIVISION ANNUAL REPORT, APRIL 1, 1977-MARCH 31, 1978. (Japan Atomic Energy Research Inst., Tokyo). Oct 1978. 215p. Dep. NTIS (US Sales Only), PC A11/MF A01.

Research activities in the Division of Reactor Engineering in fiscal 1977 are described. Works of the Division are development of multi-purpose Very High Temperature Gas Cooled Reactor, fusion reactor engineering, and development of Liquid Metal Fast Breeder Reactor for Power Reactor and Nuclear Fuel Development Corporation. Contents of the report are nuclear data and group constants, theoretical method and code development, integral experiment and analysis, shielding, heat transfer and fluid dynamics, reactor and nuclear instrumentation, dynamics analysis and control method development, fusion reactor technology, and Committee on Reactor Physics.

- 2143 (JAERI-M-8117) EXPERIMENTAL STUDY OF LMFBR HETEROGENEOUS CORE AT FCA. Nakano, M.; Iijima, S.; Shirakata, K.; Hirota, J. (Japan Atomic Energy Research Inst., Tokyo). Mar 1979. 18p. NTIS (US Sales Only), PC A03/MF A01.

In order to investigate physics properties of the heterogeneous LMFBR core and to examine reliability of the current data and method for heterogeneous core configuration, an experimental study has been made on FCA VII-3 Assembly which has an internal blanket (IB) at midplane of the cylindrical core. Systematic measurements of the criticality, sample worth, fission rate ratio, reaction rate distribution and sodium void worth were made on the heterogeneous cores whose IBs were different in

composition and thickness. The sodium void worth for the region from the midplane to the middle of the core decreases with increase of ^{235}U or Pu atomic density in the IB. The worth is, however, not sensitive to composition of the IB. The void worth in the core is lowered by about 35% when the thickness of the IB is increased from 20 cm to 40 cm. The initial analysis using the JAERI-Fast Set Version II and the diffusion code CITATION was made for Assembly VII-3-1, the core with IB of 30 cm thickness. Comparison between the calculated and experimental results revealed the following: Pu worth is not well predicted in the core. Reaction rates sensitive to low energy neutrons are underestimated in the IB when they are normalized in the core. A discontinuity of more than 20% in the C/E value of the sodium void worth exists at the interface of the IB and the core.

- 2144 (JAPFNR--406) SODIUM MIST BEHAVIOR IN COVER GAS SPACE OF AN LMFBR: EXPERIMENTAL STUDY. Himeno, Y.; Takahashi, J. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). Feb 1978. 33p. (PNC-N--941-78-42). Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

The paper describes the sodium mist behavior in a simulated Argon cover gas space of an LMFBR consisting of a test vessel of 1,400 mm in axial length, 305.5 mm in inner diameter and about 100 liters in volume. Measurements were made of the mist concentration and the mist gravitational settling flux between the sodium pool temperature range of 290° to 520°C. The results are discussed under the monosize assumption of the particles, and the particle sizes and evaporation rate are derived. Transient and steady state mist concentration behavior were also investigated.

- 2145 (JAPFNR--422) LOCAL FLOW BLOCKAGE EXPERIMENTS IN 37-PIN SODIUM COOLED BUNDLES WITH GRID SPACERS. Uotani, M.; Haga, K.; Kikuchi, Y.; Hori, M. (Power Reactor and Nuclear Fuel Development Corp., Oarai (Japan)). Oct 1978. 16p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

A series of out-of-pile experiments were conducted on local temperature rises due to non-heat generating blockages in 37-pin bundles. In the central blockage experiment, the central 24 subchannels of the bundle were blocked with a 5 mm thick stainless-steel plate at upstream end of a grid spacer. The blocked area was 27% of the total flow area. In the edge blockage experiment, a stainless-steel plate blocked 39 subchannels of a 1/2 edge part of the cross-sectional area. In both cases of the central and the edge blockage, the pin surface temperature attained a local peak in the grid spacer behind the blockage and decreased with the distance from the blockage. The mass exchange rate and the coolant residence time in the wake regions were evaluated from the observed temperature rises. The mass exchange rate per unit interface area behind the edge blockage agreed fairly well with that behind the central blockage. The dimensionless coolant residence time was found independent of Reynolds number except in the low number range, and the value obtained in the edge blockage experiment was about 2.4 times as much as that obtained in the central blockage experiment. When experimental results were extrapolated to the reactor condition, an edge blockage of more

than 30% might cause local boiling in the wake region, while a central one would not cause local boiling in any blockage ratio less than 60%. The temperature rises in the blocked grid spacer were also discussed.

- 2146 (JAPFNR--423) ACOUSTIC NOISES WITH LOSS-OF-FLOW SODIUM BOILING EXPERIMENT IN 19-PIN BUNDLE. Ozaki, Y.; Haga, K.; Kikuchi, Y.; Morikawa, T.; Hori, M. (Mitsubishi Electric Corp., Amagasaki, Hyogo (Japan). Central Research Lab.; Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). Oct 1978. 10p. Dep. NTIS, PC A02/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

The papers deals with the measurement of acoustic noises with sodium boiling in an electrically heated 19-pin bundle which simulated an LMFBR fuel subassembly. The intensity of boiling acoustic noises measured with the waveguide method was much higher than the background noises such as mechanical and flow noises. A distinct peak, which was observed at approximately 10 kHz, may be attributed to the peculiarity of sodium boiling phenomena because this peak could easily be distinguished from the resonance peaks of the experimental system. The waveform of the boiling acoustic noises was similar to the burst type acoustic emission (AE). The propagation speed of acoustic noises agreed well with a prediction by the theory based on the assumption that the measured acoustic signals were transmitted on the pipe as surface waves (Rayleigh waves) or Lamb waves.

- 2147 (JAPFNR--425) PROGRESS REPORT ON FAST BREEDER REACTOR DEVELOPMENT IN JAPAN, APRIL--JUNE 1978. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Nov 1978. 12p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

A brief summary is presented of design and development work related to the JOYO reactor and the MONJU reactor.

- 2148 (JAPFNR--434) HYDRAULIC SIMULATION OF LOCAL BLOCKAGE IN LMFBR FUEL SUBASSEMBLY. Nakamura, H.; Miyaguchi, K.; Takahashi, J. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Jan 1979. 13p. (PNC-N--941-79-07). Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

Experimental studies on the local flow blockage in LMFBR subassembly have been conducted to investigate the safety assessment. The water modeling tests have been carried out, in parallel with the development of the code UZU. Fundamental data on the flow field in a pin bundle containing a blockage have been accumulated and applied to check on uncertainties which came upon in the development of the code. The summary of the experimental results is presented.

- 2149 (JAPFNR--435) CREEP AND CREEP-RUPTURE PROPERTIES OF THE NUCLEAR FUEL CLADDING TUBE FOR FAST BREEDER REACTOR IN HIGH TEMPERATURE SODIUM: ON THE CLADDING TUBE (AISI TYPE 316SS, EFFECT OF BORON CONTENT) MANUFACTURED IN 1974 FOR MONJU. Yuhara, S.I.; Yoshida, E.; Funada, T.; Nakasuji, T. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). Nov 1978. 82p.

Dep. NTIS (US Sales Only), PC A05/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

Creep-rupture tests under internal pressure were carried out on AISI Type 316 stainless steel nuclear fuel cladding tubes for MONJU. Two kinds of nuclear fuel cladding tubes, the tubes containing 1 and 10 ppm boron respectively, were used for the creep-rupture tests, which were manufactured by the S manufacturer in the 49th year of Showa (in 1974). The nominal dimensions of these tubes are 6.5 mm in diameter and 0.45 mm in wall thickness. The Material Test Sodium Loop 1 and the Material Test Sodium Loop 2 were used for the creep-rupture tests in flowing sodium environments at the two cold trap temperatures of 190°C and 145°C, respectively. Test temperatures were set at 700° and 650°C at the test sections of the Material Test Sodium Loop 1, and at 700°C at the test section of the Material Test Sodium Loop 2.

- 2150 (JAPFNR--459) LOW-CYCLE FATIGUE TESTS ON PRIMARY COOLANT PIPING SYSTEM COMPONENTS OF FAST REACTORS. Imazu, A. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). [nd]. 31p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United State-Japanese Fast Reactor Exchange Program.

The primary objectives of this series of tests are as follows: to assess the validity of the design method provided by Section III of ASME Boiler and Pressure Vessel Code with emphasis placed on low-cycle fatigue design; to make observations of propagation and penetration behavior of fatigue cracks developed in piping components; and to collect data relating to the inspections of a fast reactor in possible use. These data consist mainly of the measurements of acoustic emission.

- 2151 (JAPFNR--472) REACTOR PHYSICS ACTIVITIES RELEVANT TO FBR AND ATR PROGRAMS IN PNC, JAPAN, NOVEMBER 1978-SEPTEMBER 1979. Inoue, T. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). 1979. 11p. (PNC-N--241-79-26). Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States-Japanese Fast Reactor Exchange Program.

Reactor physics activities have been performed in PNC, Japan, to support FBR and ATR development programs. The experimental fast reactor JOYO attained the second-stage thermal power of 75 MW in July 1979. Environmental impact of the prototype fast breeder reactor, MONJU, is now being assessed by both Japanese government and prefectural government and its construction will be started in 1981. A prototype reactor of heavy water-moderated, boiling light water-cooled, pressure tube type, FUGEN, went into its commercial operation in March 1979.

- 2152 (JAPFNR--473) ACTIVITIES IN SBL-FPL LOOPS FOR SAFETY STUDY OF LMFBR(I). (Power Reactor and Nuclear Fuel Development Corp., Darai, Ibaraki (Japan). Darai Engineering Center). Jun 1979. 21p. (PNC-N--941-79-79). Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under USDOE-PNC FBR Information Exchange.

As the first step for the safety study of LMFBR, it is necessary to understand the initial events of possible accidents in LMFBR. Local faults and sodium boiling can be considered in that area. SBL and FPL were

constructed to understand such phenomena experimentally. Two loops own jointly the dump tank and the purification loop. SIENA is the generic name to both loops. A schematic diagram of SIENA is presented.

- 2153 (JAPFNR--479) STUDY ON PLAN OF MOCK-UP CRITICAL EXPERIMENT FOR LARGE FAST REACTORS BY USING THE FCA. (STUDY ON AN AMOUNT OF PU FUEL NEEDED FOR PARTIAL MOCK-UP EXPERIMENT IN LARGE FAST REACTORS). Takano, H.; Mitani, H.; Matsui, Y.; Ishiguro, Y.; Kuroi, H. (Power Reactor and Nuclear Fuel Development Corp., Darai (Japan); Japan Atomic Energy Research Inst., Tokyo). Jul 1979. 84p. (In English and Japanese). (PNC-SJ--250-79-18; JAERI-Memo--8362). Dep. NTIS (US Sales Only), PC A05/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

Study on Pu-contents needed for partial mock-up experiment in large fast reactors was performed by considering zone loading, sector loading, and partial sum mock-up methods. Integral quantities, i.e., central reaction rate ratio, reaction rate distribution, Doppler reactivity coefficient, Na-void reactivity and control rod reactivity were calculated for full mock-up and these partial mock-up cores with various Pu-contents.

- 2154 (JAPFNR--482) OUT-OF-PILE EXPERIMENT OF FISSION GAS RELEASE IN LMFBR SUBASSEMBLY. 4. FEASIBILITY STUDY OF GAS RELEASE DETECTION BY USE OF FLOW FLUCTUATION SIGNAL. Irushima, H.; Ogino, T.; Ohkouchi, T.; Komaba, T.; Haga, K.; Hori, M. (Power Reactor and Nuclear Fuel Development Corp., Darai, Ibaraki (Japan). Darai Engineering Center; Mitsubishi Electric Corp., Amagasaki, Hyogo (Japan). Central Research Lab.). May 1979. 34p. (In English and Japanese). (PNC-N--941-79-49). Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States-Japanese Fast Reactor Exchange Program.

Fission gas release tests were carried out in a 37-pin bundle to study the feasibility of fission product gas release detection by use of eddy current type flow meter. This 37-pin bundle simulated a normal subassembly of LMFBR. Flow and flow fluctuations were measured by an eddy-current type flow meter at the outlet of the bundle. The output signals from the flow meter were filtered and amplified through a specially made fluctuation measuring circuit. Thus, these measured signals were analyzed using a digital computer.

- 2155 (JAPFNR--487) ASSESSMENT OF GENERATION METHOD OF NUCLEAR REACTOR CONSTANTS SET USING THE JENDL-2 NUCLEAR DATA FILE. Takano, H.; Nakagawa, M.; Hasegawa, A.; Ishiguro, Y.; Matsui, Y.; Kaneko, K. (Japan Atomic Energy Research Inst., Tokyo). 1979. 67p. (JAERI-Memo--8233; PNC-J--250-79-13). Dep. NTIS (US Sales Only), PC A04/MF A01.

Work performed under United States-Japanese Fast Reactor Exchange Program.

Interpolation method of resonance shielding factor, group collapsing method with bilinear weight, and group structure were assessed for the purpose of generation of new nuclear reactor constants set using the JENDL-2 nuclear data file, and an adjustment plan was proposed for the methodology used. Concerned with the interpolation method of shielding factor, various methods used so far and, additionally, spline fitting method and Akima's method were intercompared in connection with computing accuracy and time. The spline method was found to give the most excellent and stable accuracy.

Concerning the bilinear-weight collapsing method, the difference between the flux- and bilinear-weight collapsings was studied by considering a group collapse from seventeen to twenty-five groups and by analyzing reactivity worth measurements for sodium void at ZPPR-2 and steam entry at ZPPR-9 Phase 1. The flux-weight collapsing method used so far was shown to be sufficient for the analysis of the measurement of sodium void reactivity worth.

- 2156 (JAPFNR--488) INVESTIGATION ON NEUTRONIC PERFORMANCES OF LARGE SCALE HETEROGENEOUS LMFBR. Iijima, S.; Yoshida, H.; Hotta, M. (Power Reactor and Nuclear Fuel Development Corp., Daiichi (Japan); Japan Atomic Energy Research Inst., Tokyo). Apr 1979. 32p. (In English and Japanese). (JAERI-Memo--8266; PNC-J--250-79-14). Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

Heterogeneous LMFBR concept with internal blanket has received substantial attention for its good breeding performance and fairly low sodium void reactivity effect. A comparative study of neutronic performances between conventional and heterogeneous LMFBRs has been made to identify what is the main issue in analyzing physics performance of heterogeneous LMFBR. The heterogeneous LMFBRs investigated in the study were Axial Parfait and Radial Heterogeneous LMFBRs with power 1000 MWe and the neutronic performances were burnup, breeding, power distribution, and sodium void reactivity effect. Uncertainty of the neutronic performances related to neutron cross-section data uncertainty was also examined in the study using three different 25-group cross-section sets generated from JAERI-FAST, ENDF/B-IV and JENDL-1.

- 2157 (JAPFNR--494) STUDIES OF RARE EARTH ELEMENT AS CONTROL ROD MATERIAL FOR LMFBR. PART V. Suzuki, H.; Iseki, T.; Maruyama, T.; Ohno, A. (Tokyo Inst. of Tech. (Japan). Research Lab. of Nuclear Reactor). Apr 1979. 37p. (In Japanese). (PNC-J--261-79-02). Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

Compatibility of Eu_2O_3 and a mixed oxide of $(\text{Eu,Gd})_2\text{O}_3$ with Type 316 stainless steel was examined by heating sintered monoclinic Eu_2O_3 pellets and the mixed oxide of $(\text{Eu,Gd})_2\text{O}_3$ pellets in contact with Type 316 stainless steel spacers. The heat-treatment was carried out in the range from 550°C to 1000°C for 500 h to 5000 h in out-of-pile conditions. The density of both Eu_2O_3 pellet and $(\text{Eu,Gd})_2\text{O}_3$ pellet is approximately 95% of theoretical density. The ratio of number of Eu to Gd atoms contained in the mixed oxide of $(\text{Eu,Gd})_2\text{O}_3$ is adjusted to be 1.0. Metallographic examination of the heat-treated stainless steel spacer indicated that chemical reaction between Eu_2O_3 and Type 316 stainless steel occurred at 650°C, and severe reaction, particularly grain boundary penetration, was observed in stainless steel sample which was heat-treated at 1000°C for 500 h.

- 2158 (JEN--363) CREEP-RUPTURE-STRENGTH AND CREEP-BEHAVIOUR OF STAINLESS STEEL X6CRNI 1811 (DIN 1.4948). Solano, R.R.; de las Rivas, M.; Seith, B.; Schirra, M. (Junta de Energia Nuclear, Madrid (Spain)). 1977. 25p. (In Spanish). Dep. NTIS (US Sales Only), PC A02/MF A01.

The steel X6CrNi 1811 (DIN 1.4948) that will be used as a structure material for the German

fast breeder reactor SNR 300, was creep-tested in a temperature range of 550-650 deg C under base material condition as well as welded material condition. Tests are foreseen up to 30,000 hours with a continuous measuring of the elongation. The test results up to about 4,000 hours are described. Taking into account the results of other programs carried out with the same material between 550 and 600 deg C at similar rupture time, the stresses were defined for the longterm test. The main point of this program ('Extrapolation Program') lies in the knowledge of the creep-rupture-strength and creep behaviour of the structure materials up to 3.10^4 h at high temperature in order to extrapolate up to 10^5 h for reactor operating temperatures.

- 2159 (JEN--389) PHYSICAL PROPERTIES OF LIQUID SODIUM. Alberdi Priniccia, J.; Martinez Piquer, T.A. (Junta de Energia Nuclear, Madrid (Spain)). 1977. 35p. (In Spanish). Dep. NTIS (US Sales Only), PC A03/MF A01.

Molten sodium has been the more accepted coolant for the first generation of FBR; for this reason the knowledge of its technology is needed for the development of the next LMFBR. A series of data necessary for designing liquid sodium systems is given. Tables and graphs about the most important physical properties of sodium between 1200-1400°C are gathered. The results have been obtained from equations that relate the properties with temperature using a Fortran IV program.

- 2160 (JEN--447) CREEP-RUPTURE-TESTS ON THE STAINLESS STEEL X6 CRNI 1811 (DIN 1.4948) IN THE FRAME OF THE 'EXTRAPOLATION-PROGRAM' PT. 2. Solano, R.R.; Barroso, S.; de las Rivas, M.; Schirra, M.; Seith, B. (Junta de Energia Nuclear, Madrid (Spain)). 1979. 58p. (In Spanish). NTIS (US Sales Only), PC A04/MF A01.

The austenitic stainless steel X6 CrNi 1811 (DIN 1.4948) that is used as a structure material for the German Fast Breeder Reactor SNR 300 was creep tested in a temperature range of 550-650 deg under base material condition as well as welded material condition. The main point of this program ('Extrapolation-Program') lies in the knowledge of the creep-rupture-strength and creep-behaviour up to 3X10⁴ hours at higher temperatures in order to extrapolate up to $(\geq)10^5$ hours for operating temperatures. In order to study the stress dependency of the minimum creep rate additional tests were carried out over temperature range 550 deg - 750 deg C. The present report describes the state in the running program with test-time up to 35,000 hours. Besides the creep-rupture behaviour it is possible to make a distinct quantitative statement for the creep-behaviour and ductility. Extensive metallographic examinations show the fracture behaviour and changes in structure.

- 2161 (KFK--1276/4) FOURTH QUARTERLY REPORT 1976. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueker). Apr 1976. 204p. (In German).

The 4th quarterly report 1976 describes activities which were carried out within the framework of the Fast Breeder project at the Gesellschaft fuer Kernforschung, Karlsruhe, or on its behalf. The sections refer to the following subjects: Fuel rod development; material studies and development; corrosion experiments and coolant analysis; physics experiments; reactor theory; fast breeder safety; instrumentation and signal processing for core control; effects on the environment;

sodium technology experiments; thermo- and fluid-dynamic tests in gas; design studies concerning sodium-cooled breeders.

- 2162 (KFK--1277/2, pp v) REACTOR THEORY AND PHYSICAL EXPERIMENTS. Jiang, S.H.; Werle, H.; Froehner, F.H.; Stein, E.; Krieg, B.; Fischer, E.A.; Kiesel, R.; Brandl, V.; Hoebel, W.; Kuefner, K. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Oct 1977. (In German).
Fast Breeder Project: Second quarter annual report 1977.

The neutron induced gamma radiation in iron has been determined in a test rig using fission neutrons of Cf-252. In the range between 0.5 and 2.0 MeV neutron energy the agreement with results obtained by gamma transport calculations with KEDAK data is satisfactory; below this region there are major deviations. Cross sections for resonance capture in structural materials (Fe, Ni, Cr) in the range of up to 0.3 MeV have been newly evaluated. This also allows a more accurate prediction to be made of the influence of these materials on the nuclear Doppler coefficient. To back up calculations by transport methods results of the SNEAK 9 C 2 critical assembly have been recalculated by Monte-Carlo methods. The results obtained by both techniques largely agree, but both differ considerably from the measured results, which is probably due to the inaccurate knowledge of cross sections. A summary is given of the use, further development and data management of the KAPROS reactor program system.

- 2163 (KFK--1277/2, pp v) REACTOR TECHNOLOGY. Wild, E.; Mack, K.J.; Dres, K.; Seeger, W.; Trippe, G.; Weinberg, D.; Marten, K.; Wiens, E.; Hoffmann, H. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Reaktorbauelemente); Rehme, K. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Oct 1977. (In German).
Fast Breeder Project - Second quarter annual report 1977.

The pressure drop, flow distribution, levitation point and subassembly stability were studied on a precursor model of a series of absorber bundles of the first shutdown unit of the SNR 300. Temperature form factors as a function of radial power form factors were determined by means of thermohydraulic calculations for various types of spacers of core elements. According to these calculations, helical wire spacers result in the most pronounced coolant temperature variations over the cross section of the element, and in the most adverse mean coolant temperature as well.

- 2164 (KFK--2353) TUFCI, AN EXPERIMENTAL FACILITY TO STUDY THE MOLTEN-FUEL-COOLANT-INTERACTION IN A FULL SCALE FUEL ELEMENT. Beutel, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). Feb 1977. 66p. (In German). Dep. NTIS (US Sales Only), PC A04/MF A01.

The safety-investigations for the Liquid-Metal-Fast-Breeder-Reactor also include experiments for fuel-coolant-interactions (FCI) in a simulated fuel element to find out the conversion factor from thermal to mechanical energy. This report describes the development of an experimental facility which makes it possible to produce an FCI in a geometry, which is typical for a fuel element. Other than in

real elements the wall of the reaction tube is rigid. Fuel-sodium contact is achieved by the impact of sodium and molten UO_2 as well as by the injection of smaller quantities of sodium into UO_2 (initiating-reaction). Finally an experiment is described in which smaller quantities of sodium were injected into liquid fuel.

- 2165 (KFK--2356) EXPERIMENTAL DETERMINATION OF TEMPERATURE FIELDS IN SODIUM-COOLED ROD BUNDLES WITH HEXAGONAL ROD ARRANGEMENT AND GRID SPACERS. Moeller, R.; Tschoeke, H.; Kolodziej, M. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorbauelemente; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). Jan 1977. 46p. (In German). Dep. NTIS (US Sales Only), PC A03/MF A01.

Three-dimensional temperature fields in the claddings of sodium cooled rods were determined experimentally under representative nominal operating conditions for a SNR typical 19-rod bundle model provided with spark-eroded spacers. These experiments are required to verify thermohydraulic computer programs which will provide the output data for strength calculations of the high loaded cladding tubes. In this work the essentials are reported of the measured circumferential distributions of wall temperatures of peripheral rods. In addition the sub-channel temperatures measured over the bundle cross section are indicated, they are required to sustain codes for the global thermohydraulic design of core elements. The most important results are: 1) The whole fuel element is located within the thermal entrance length. 2) High azimuthal temperature differences were measured in the claddings of peripheral rods, which are strongly influenced by the distance between the rod and the shroud, especially for the corner rod. 3) With decreasing Pe-number (< approximately 360) the circumferential temperature differences increase at a superproportional rate. The results of experiments described call for the settlement with priority of the following problems: 1) Experiments to determine the influence of the distance between the rods and the shroud on the temperature distribution in the claddings of peripheral rods, above all corner rods. 2) Completion of the presently used global design codes by multi-zone models applicable for local calculations of critical bundle zones.

- 2166 (KFK--2360) EXPERIMENTAL INVESTIGATIONS INTO COOLANT CROSS MIXING IN MULTI-ROD-BUNDLES WITH GRID-TYPE SPACERS IN TURBULENTLY FLOWING SODIUM. Hoffmann, H.; Baumgaertner, E.; Frey, H.H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorbauelemente; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). Jan 1977. 17p. (In German). Dep. NTIS (US Sales Only), PC A02/MF A01.

The thermo- und fluiddynamic design of fuel elements of Liquid Metal-Cooled Fast Breeder Reactors needs characteristic values for the magnitudes of the coolant cross mixing. These were determined experimentally in sodium flow. 61-rod clusters were used with the specific dimensions of the SNR-300 fuel element (rod pitch to diameter ratio $P/D = 1.32$) equipped with grid-type spacers. The investigations were performed using the 'heating method'. The results show that the mixing rates are mainly influenced by the spacer type and the P/D -ratio.

- 2167 (KFK--2368) HEATER RODS FOR SIMULATION OF FUEL PINS IN LIQUID-METAL-BOILING EXPERIMENTS. PT. 1. Casal, V.; Demski, A.; Graf, E.; Hartmann, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorbaulemente; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). Jan 1977. 25p. (In German). Dep. NTIS (US Sales Only), PC A02/MF A01.

The report describes the design, development, and testing of electrical heater rods for boiling tests in LMFBR fuel element simulators. The tests are required as a proof of the capability to control local and gross boiling in the core of SNR-300. The setup and performance of the tests call for heater rods simulating with a very high operating reliability the assumed specifications of SNR-300, even under transient conditions. Loads of the heating surface of up to 170 W/cm² must be attained at coolant temperatures of 950°C.

- 2168 (KFK--2387(Pt.7)) KEDAK PROGRAM COMPENDIUM. PART VII: CALCUL-CALCULATION. Langner, I.; Meyer, R. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Nov 1977. 121p. Dep. NTIS (US Sales Only), PC A06/MF A01.

Work performed under United States--West Germany Fast Reactor Exchange Program.

CALCUL is a program system to calculate composed cross section quantities and to arrange them according to KEDAK-conventions. A special command language has been generated to simplify the input and to provide interactive communication under time-sharing-option. In CALCUL any amount of data can be processed. The output of the program is a data set which can be directly used as an input to program system KEMA which modifies the KEDAK-library.

- 2169 (KFK--2416, pp 27-37) CORE ELEMENTS OF THE SNR-300. Biermann, W.; Toebbe, H.; Wenk, K.H. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). 1976. (In German).

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. (28 - 29 Oct 1976).

A survey is presented of the various core element types of the SNR core and of their constructional design.

- 2170 (KFK--2416, pp 80-88) STATUS OF DEVELOPMENT OF CLADDING AND STRUCTURAL MATERIALS FOR THE SNR CORE ELEMENTS. Dietz, W. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). 1976. (In German).

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Problems concerning high-temperature plasticity and the swelling of cladding and fuel channel materials of the SNR-300 are dealt with. Plasticity of the material 1.4970 under short-term and long-term stress (10⁴h), the 0.2% yield point, and elongation before reduction of area of the irradiated material as well as creep rupture strength and the 0.2% creep limit of pipes made of 1.4970 (t = 700°C, irradiated and non-irradiated) are presented in diagrammes. Creep induced by irradiation of the materials 1.4970 and 1.4981 is also dealt with.

- Funke, P. (Alpha Chemie und Metallurgie G.m.b.H. (ALKEM), Hanau (Germany, F.R.)). 1976. (In German).

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

A survey is presented of the fabrication of oxide and carbide fuels and of the fuel rod for fast breeders (KNK, SNR-300). The advantages of the chosen methods are explained. The main points of development concerning the oxide fuel rod are gone into. The process sequence for plutonium oxide and plutonium carbide processing is presented in a flow chart.

- 2172 (KFK--2416, pp 4-11) BASIC IDEAS ON THE DEVELOPMENT OF FUEL ELEMENTS FOR FAST BREEDER REACTORS. Karsten, G. 1976. (In German).

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Within the research for the development of fuel elements for sodium-cooled fast breeders, the emphasis is shifted from strategic to basic matters. Just breeding gain and burn-up are no longer to the fore, but fabrication methods and material problems above all. With regard to fabrication methods, shielding against plutonium and americium by means of mechanisation and automation plays a leading role, with regard to reprocessing it is fuel solubility. Concerning materials problems, the issue is resistivity against radiation damage, i.e. embrittlement, swelling and creep, and, in the accident programme, unsteady loads. Carbide and burn-up research, currently running second, is concerned with questions related to fabrication and concepts, and in the case of burn-up research with reliability tests.

- 2173 (KFK--2416, pp 19-26) FUEL ELEMENTS FOR BREEDER POWER PLANTS. INTERNATIONAL STATUS. Kummerer, K. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). 1976. (In German).

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

A survey is presented of 21 fast breeder reactors which are either already in operation, under construction, or in a concrete stage of planning. Design and operational data of the fuel rods and fuel elements of these reactors are given in tabular form. A synopsis reveals common aspects and differences in the line of development of the various countries.

- 2174 (KFK--2416, pp 60-70) STATUS OF THE DEVELOPMENT OF THE CARBIDIC FUEL ELEMENT. Muehling, G. 1976. (In German).

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

The carbide irradiation programme for developing carbide fuel elements for the SNR-300, one of the programme items of the Gesellschaft fuer Kernforschung, Karlsruhe, is introduced. It is subdivided into 2 phases: in phase 1, individual rod irradiation is carried out in the thermal reactors FR-2 and BR-2. Phase 2 includes the focal points: cluster irradiation under rapid flux and tests concerning the incident behaviour of carbide rods. Irradiations carried out so far demonstrate that the swelling of fuel and can be got under control if the design is right.

- 2171 (KFK--2416, pp 96-101) TECHNOLOGY OF THE PRODUCTION OF BREEDER FUEL ELEMENTS.

- 2175 (KFK--2416, pp 38-41) FIRST DRAFTS OF CORE AND FUEL ELEMENTS FOR THE SNR-2 REACTOR. Spenke, H.; Wehmann, U. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). 1976. (In German).

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

FUEL RODS;OPTIMIZATION;PLANNING;REACTOR CORES;SNR-2 REACTOR

- 2176 (KFK--2416, pp 102-108) DESIGN OF CARBIDE FUEL RODS. Steiner, H. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). 1976. (In German).

From 5. space optics meeting; Marseille, France (14 Oct 1975).

The model theory for the design of carbide fuel rods, unlike that of the oxide fuel rod, is considered to be not yet fully developed. Therefore, one falls back on empirical material from irradiation experiments. The paper is concerned with thermal design, the swelling of carbide fuel, and the mechanical interactions of the system fuel/can. The following problems are dealt with in this connection: The space between fuel and can as a heat barrier, the temperature drop in the carbide pellet, model calculations on heat transfer, a rupture study of the carbide pellet, the stretching of cans during irradiation experiments, integral swelling rates, swelling pressure and creep deformation of cans.

- 2177 (KFK--2416, pp 72-79) CHEMICAL AND MECHANICAL INTERACTION IN THE OXIDE FUEL ROD. Thuemmler, F.; Dienst, W. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). 1976. (In German).

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

A survey is presented of chemical and mechanical interactions between mixed oxide fuel and can material, whereby the problems of fuel rod behaviour under load changes are dealt with in more detail. It is found that in all cases of cladding tube cracking internal corrosion plays a role.

- 2178 (KFK--2416, pp 50-59) DETECTIVE ROD AND ACCIDENT EXPERIMENTS. Plitz, H. (Kernforschungszentrum Karlsruhe (Germany, F.R.)). 1976. (In German).

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Fuel element behaviour of the SNR-300 during possible operational incidents is of particular interest for reasons of economy. Programmes for investigating the effects of operational incidents on the behaviour of fuel elements, and the behaviour of defective fuel rods during continued operation, are therefore carried out. The experimental incident programmes (performance and coolant incidents, cyclic operation, overload) and the programme concerning defective rods are introduced. The problem in connection with the Na-fuel swelling process is dealt with.

- 2179 (KFK--2416, pp 42-49) CORE ELEMENTS DURING RAMP POWERS AND INCIDENTS. Toebe, H.; Gestermann, F. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). 1976. (In German).

From KTG technical day on fuel element of

the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Three modes of operation are taken as the basis of design for the core elements of the SNR 300, from which dissimilar design criteria are derived. Fuel rod behaviour during reactivity and coolant throughput is dealt with. Three examples of incidents, the loads during which and their effect on the core elements are explained and assessed, demonstrate that the core elements meet with the design requirements in cases of typical incidents.

- 2180 (KFK--2416) FUEL ELEMENT OF THE LMFBR. Karsten, G. (ed.). (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). 1976. vp. (In German). (CONF-7610135--). NTIS.

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Separate abstracts are included for each of the individual papers presented.

- 2181 (KFK--2416, pp 89-95) DEVELOPMENT OF CLADDING AND STRUCTURAL MATERIALS. Anderko, K. 1976. (In German).

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

The creep-fracture behaviour of the irradiated material 1.4970 and possibilities to improve its behaviour are discussed. From this, it follows that the alloy 1.4970 ought to be strengthened more still with regard to creep rupture ductility and strength. The planned optimization programme considers the following parameter variations: 1. Solution heat temperature; 2. cooling rate; 3. thermo-mechanical treatment; 4. chemical modification (B, C, Ti).

- 2182 (KFK--2416, pp 12-18) SNR-CORES: KNK-II, SNR-300 UP TO SNR-2. Hoechel, J. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). 1976. (In German).

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

The focal points in the development of fast breeder core design are outlined briefly, and the main data of the cores Na-2, SNR-300 MK-I, SNR-300 MK-Ia and SNR-300 MK-II are presented in tabular form. Steel 1.4970 is named as a new reference material for SNR fuel elements.

- 2183 (KFK--2451) LAYOUT, DESIGN AND POST IRRADIATION EXAMINATION OF NA-BONDED FUEL PINS OF THE TEST GROUP MOL 15. Steiner, H.; Freund, D.; Jacobi, O.; Weimar, P. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). Apr 1977. 80p. (In German). Dep. NTIS (US Sales Only), PC A05/MF A01.

The experiment Mol 15 includes the irradiation of 4 fuel rod specimens with UC/PuC fuel (25 wt% Pu and Usb(nat)) of different M_2C_3 shares with Na bonding. It is the first capsule irradiation of Na-bonded carbide rods within the Project Fast Breeder. In the present report, design, irradiation, and post-irradiation examination of the fuel rods are described. At linear power ratings of maximum 1,150 W/cm and cladding temperatures of maximum 500°C burn-ups of nearly 9% fima were obtained without getting rod defects. Cladding deformations remained decidedly small (0.3% at

the maximum). The cladding consisted of the material 1.4988 (X8CrNiMoVNB1613). The fuel structure is characterized by a highly porous zone in the interior followed by a narrow high-density zone. On three specimens, indications of a central cavity were found. Together with the anomalous bubble size, this suggests superheating of the after failure of the Na bonding during irradiation.

- 2184 (KFK--2472) HEAT TRANSMISSION DURING STARTUP IN CARBIDE FUEL RODS PROVIDED WITH AVERAGE TO LARGE FABRICATION GAPS. Steiner, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). Jun 1977. 14p. (In German). Dep. NTIS (US Sales Only), PC A02/MF A01.

The influence was investigated of different geometry effects on the temperature distribution in helium bonded carbide fuel pins. For this purpose the heat conduction equation was solved using the development of the Fourier-series. Models were set up to describe concentric, excentric and stacked arrangements of pellets not cracked as well as pellet cracking followed by the displacement of pellet fractions towards the cladding. It appeared that the heat transmission coefficients, determined experimentally during startup, can be explained solely by the latter effect. The fact that this effect has not been taken into account is the cause of the discrepancy between the experimental results and the predictions by the GAPRS heat transmission model based on a concentric pellet not cracked.

- 2185 (KFK--2474) HEATER RODS FOR SIMULATION OF FUEL PINS IN LIQUID-METAL-BOILING EXPERIMENTS. Pt. 2. Casel, V.; Demski, A.; Graf, E.; Hartmann, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). Aug 1977. 25p. (In German). Dep. NTIS (US Sales Only), PC A03/MF A01.

This report describes advancements in the development of electric heater rods to be used in sodium boiling experiments. The heat is generated in the heater rod by ohmic heating of a tape helix. The heat passes through the insulator, a layer made of compacted boron nitride powder, and reaches the clad of the heater rod. By use of tantalum cladding tubes and conductors consisting of tantalum 10% tungsten, heat surface loads of more than 250 W/cm² can be realized with these heater rods at clad temperatures of 1050°C. Since the geometric layout and the fabrication methods were maintained, it was possible to benefit from experience gathered on heater rods for boiling experiments equipped with nickel-chromium conductors and stainless steel clads, which have operated successfully.

- 2186 (KFK--2475) DESCRIPTION OF THE TESTSECTIONS FOR LOCAL BOILING EXPERIMENTS IN THE KNS-LOOP. Huber, F.; Jenes, E.; Mattes, K. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). Apr 1978. 19p. (In German). Dep. NTIS (US Sales Only), PC A05/MF A01.

Within the context of the SNR-300 R+D program experimental studies have been made on the effects of local cooling disturbances within large pin bundle geometry, under conditions similar to those in the reactor. To this two test sections were built which were geometrically and thermohydraulically similar

to a SNR-300 fuel element. The first test section possesses a central 49% blockage and the second a 21% edge blockage. This report is a description of both the test section and considers the construction and instrumentation in particular detail.

- 2187 (KFK--2490) HYPOTHETICAL CORE DISRUPTIVE ACCIDENT ANALYSIS OF A 2000 MWSUB(E) LIQUID METAL COOLED FAST BREEDER REACTOR. Struwe, D. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). Dec 1977. 170p. (In German). Dep. NTIS (US Sales Only), PC A09/MF A01.

A structural phase diagram for hypothetical core disruptive accidents (HCDA) has been developed based on a variety of analyses for different LMFBR's. The intention was to identify the strategic phases of HCDA's important with regard to safety aspects of the plant. These phases are investigated in detail for a 2,000 MWSUB(e) LMFBR (SNR=2,000). Characteristic data of SNR=2,000 are discussed concerning their influence on safety analysis. Reasons for the choice of model parameters for special phenomena as fuel coolant interaction, fuel pin failure mechanisms and sodium voiding are given. The results of calculations with CAPRI-2, HOPE and KADIS are analyzed for possibilities to enter energetic core disassembly with consequences, making power values below 2,000 MWSUB(e) necessary. Investigation of these results shows that the expected consequences do not lead to design requirements, restricting the magnitude of the electrical power output of LMFBR's to values below 2,000 MWSUB(e). Therefore, consequences of HCDA's are principal not expected to limit the feasibility of conventional core design of this order of magnitude.

- 2188 (KFK--2491) ANALYSIS OF THE DISASSEMBLY PHASE OF HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS (HCDA) FOR THE SNR-2000. Maschek, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). Dec 1977. 245p. (In German). Dep. NTIS (US Sales Only), PC A12/MF A01.

In the present work main emphasis has been layed on the analysis of the disassemblyphase of HCDAs. In the first chapters the essential core design features and some remarks about the disassembly analysis of very large conventional breeder cores have been discussed. The course of the fore-mentioned accident types has been followed up until first neutronic shut down with the Karlsruher disassembly code KADIS. For all accidents the power histories, the energy and mass of molten fuel - which are measures for the severity of the excursion - are displayed. In a first approach the mechanical loads of the reactor tank structures generated by the power excursions have been estimated. Additionally the influence on the development of the loss of flow accident when using an extreme slow pump coast down curve or providing the subassembly inlet with flow diodes with direction dependent orificing properties, have been analyzed. The results of the safety analysis strongly support the conclusion that the requirements to withstand the consequences of HCDA's in principle will not limit the feasibility of conventionally designed LMFBRs up to a power of 2000 MW electric.

- 2189 (KFK--2497) KADIS: A PROGRAM TO ANALYSE THE DISASSEMBLY PHASE OF HYPOTHETICAL ACCIDENTS IN LMFBRs. Schumuck, P.; Jacobs, G.;

Arnecke, G. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueiter). Nov 1977. 218p. (In German). Dep. NTIS (US Sales Only), PC A11/MF A01.

The program KADIS models the disassembly phase during power excursions in LMFBR hypothetical accidents. KADIS is based on point kinetics in the neutronics part and on a 2-dimensional representation of the reactor core in the hydrodynamics part. The core is modeled as an ideal, compressible fluid which is heated up adiabatically during the excursion. KADIS was built up with the help of the VENUS program of Argonne National Laboratory. Several important features were added to the basic VENUS model. Therefore we give first a complete description of the mathematical models used. Secondly we provide the user with the necessary information to handle the input/output of KADIS.

2190 (KFK--2508) KINTIC-2 USER'S MANUAL. Vaeth, L. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueiter). Nov 1977. 99p. Dep. NTIS (US Sales Only), PC A06/MF A01.

KINTIC-2 is a two dimensional reactor dynamics program, which can be used to analyze operational reactor transients or transients caused by accident situations. In the near future, it will be mainly used for the calculation of the predisassembly phase of LMFBR hypothetical accidents. Reactor kinetics may optionally be calculated with a simple point kinetics model or the adiabatic, normal or improved quasistatic method. The thermodynamics and material motion part, which was adapted from the CAPRI-2 system, uses representative coolant channels in a single channel model. It treats various phenomena which occur in intact fuel subassemblies: One phase steady state and transient heat transfer and coolant pressure distribution, pin deformation and failure, sodium voiding, slumping, and fuel coolant interaction. The report gives a short summary of the underlying physical models and some information on code organization and geometrical and thermohydraulic representation of the reactor. The main part consists of input and output description and sample cases. KINTIC-2 works in the KAPROS system and is operational since mid-1976.

2191 (KFK--2546) FURTHER DEVELOPMENT OF THE LMFBR ACCIDENT ANALYSIS CODES HOPE AND KADIS: AUTOMATIC DATA TRANSFER, NEW EQUATION OF STATE. Fischer, E.A.; Arnecke, G. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueiter). Mar 1978. 75p. Dep. NTIS (US Sales Only), PC A05/MF A01.

To provide suitable tools for the analysis of hypothetical unprotected transient overpower accidents in LMFBR, the codes HOPE and KADIS were further developed. HOPE analyses the initiating phase of a reactivity ramp accident in an irradiated core, KADIS is a core disassembly code. An automatic data transfer from HOPE to KADIS was organized. A new equation of state, which includes fission gas pressures, was developed and introduced in KADIS. The system of the two codes allows a consistent treatment of fission gas effects in both the initiating and the disassembly phase. Care was taken to make sure that the fuel-coolant interaction models in both codes are as

well compatible as feasible. A reference transient overpower accident, initiated by a 15/sec reactivity ramp, was analyzed, using a melt fraction criterion for pin failure. To make sure that an energetic core disassembly occurs, the rather pessimistic assumption of mid-plane pin failure was made. In this accident, the fission-gas driven fuel ejection through the rip causes a positive fuel motion reactivity. On the other hand, fission-gas pressure disperses fuel during the super-prompt critical disassembly phase, and the total energy release is still less than predicted in an earlier analysis of the same accident sequence with the codes CAPRI-2 and KADIS. In addition, the 15/sec ramp accident was also analyzed with the more mechanistic burst stress failure criterion, but still sticking to the pessimistic assumption of mid-plane failure. In this case, the predicted energy release is again considerably less.

2192 (KFK--2559) DETECTION OF BAND-LIMITED COMPONENTS IN NOISE SIGNALS BY SURVEILLANCE OF THE AUTO- AND CROSS POWER SPECTRAL DENSITY. Ehrhardt, J. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). Mar 1978. 115p. (In German). Dep. NTIS (US Sales Only), PC A07/MF A01.

In technical plants disturbances are possible which generate small narrow-band components in the measured signals. For the detection of these oscillatory signals the observation of the frequency spectra of the measured signals' noise components is particularly advantageous because they produce additional characteristic peaks. In this paper detection systems for the fast detection of suddenly appearing peaks in the power spectral density of noise signals are presented based on auto- and cross correlation techniques. General criteria were developed to enable the determination of the optimal detection system and its sensitivity for the different problems. Specially auto- and cross correlation measurements were compared in dependence of the signal coherence. Theoretical results were confirmed in a number of experiments. Special experimental and theoretical parameter studies were done for the optimal detection of sodium boiling in LMFBR's. Computations showed that local sodium boiling can be detected in nearly the whole core range of SNR=300 by detection systems based on the observation of the neutron flux fluctuations.

2193 (KFK--2573) PHYSICS INVESTIGATIONS OF A COMPACT SIMULATION OF A LARGE FAST BREEDER REACTOR. SNEAK-ASSEMBLY 10. Giese, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Jun 1978. 114p. Dep. NTIS (US Sales Only), PC A06/MF A01.

SNEAK-10 was the first SNEAK assembly devoted to the investigation of certain neutronic properties of breeder reactors with a nominal power of approx. 1200 MW/sub el/. The investigations were concentrated on flux tilts and related mutual control rod interactions, occurring upon changes of control insertion depths. As the physical size and fuel inventory of such reactors significantly exceed the capacity of SNEAK the investigated cores had to be compacted by a factor of approximately 2, without affecting the phenomena of interest. This was achieved by the use of densified compositions without sodium and adequate reduction of the physical dimensions. Lacking sufficient amounts of plutonium, uranium fuel was used in all investigated configurations.

- 2194 (KFK--2577) DESIGN, IRRADIATION AND POST-IRRADIATION EXAMINATION OF THE (U,Pu)C FUEL PINS OF THE TEST GROUPS 6A AND 6C. Steiner, H.; Weimar, P. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). Mar 1978. vp. (In German). Dep. NTIS (US Sales Only), PC A10/MF A01.

The documentation in this report presents the objectives of the experiment, a detailed description of design, specification and fabrication of the pins, the irradiation history, and the post-irradiation examination. In conclusion several irradiation aspects of the fuel pins are analyzed by means of computer codes.

- 2195 (KFK--2584) THEORETICAL AND EXPERIMENTAL INVESTIGATION OF THE NONLINEAR STRUCTURAL DYNAMICS OF FAST BREEDER REACTOR FUEL ELEMENTS. Liebe, R. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter; Karlsruhe Univ. (TH) (Germany, F.R.). Fakultat fuer Maschinenbau). Apr 1978. 146p. (In German). Dep. NTIS (US Sales Only), PC A08/MF A01.

Thesis.

This study describes theoretical and experimental investigations of the dynamic deformation behavior of single and clustered fuel elements under local fault conditions in a Fast Breeder Reactor core. In particular an energetic molten-fuel-coolant-interaction (FCI) is assumed in one subassembly with corresponding pressure pulses, which may rupture the wrapper and load the adjacent fuel elements impulsively. Associated coherent structural deformation may exceed tolerable and damage the control rods. To attack the outlined coupled fluid-structure-interaction problem it is assumed, that the loading at the structures is known in space and time, and that there is no feedback from the deformation response. Then current FCI-knowledge and experience from underwater core model explosion tests is utilized to estimate upper limits of relevant pulse characteristics. As a first step the static carrying capacity of the rigid-plastic hexagonal wrapper tube is calculated using the methods of limit analysis. Then for a general dynamic simulation of the complete elastoplastic subassembly response the concept of a discrete nonlinear hinge is introduced. A corresponding physical lumped parameter hinge model is presented, and general equations of motion are derived using D'Alembert's principle. Application to the static and dynamic analysis of a single complete fuel element includes the semiempirical modelling of the fuel-pin bundle by a homogeneous compressible medium. Most important conclusions are concerning the capability of the theoretical models, the failure modes and threshold load levels of single as well as clustered SNR-300 fuel elements and the safety relevant finding, that only limited deformations are found in the first row around the incident element. This shows in agreement with explosion test results that the structured and closely spaced fuel elements constitute an effective, inherent barrier against extreme dynamic loadings.

- 2196 (KFK--2586) SUMMARY OF RESULTS FOR THE SNEAK-9 SERIES OF CRITICAL EXPERIMENTS AND CONCLUSIONS FOR THE ACCURACY OF PREDICTED PHYSICS PARAMETERS OF THE SNR-300. Helm, F. (comp.). (Kernforschungszentrum Karlsruhe

G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Aug 1978. 59p. Dep. NTIS, PC A04/MF/A01.

Work performed under United States--West Germany Fast Reactor Exchange Program.

In a series of critical assemblies in SNEAK physics parameters of interest for the prototype fast reactor SNR-300 were investigated and compared to the results of calculations. Since a complete mock-up of the SNR-300 was not possible with the material supply available, the measurements were performed in three different assemblies, each being adapted to the investigation of a particular set of problems. Work was concentrated on the following quantities: criticality, breeding ratio, Na-void effect, control rod worths, and power distribution.

- 2197 (KFK--2589) NEUTRON NOISE ANALYSIS FOR MALFUNCTION DIAGNOSIS AT SODIUM COOLED REACTORS. Hoppe, P. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter; Karlsruhe Univ. (TH) (Germany, F.R.). Fakultat fuer Maschinenbau). Sep 1978. 145p. (In German). Dep. NTIS (US Sales Only), PC A08/MF A01.

Thesis.

For the investigation of the potential use of neutron noise analysis at sodium cooled power reactors, measurements have been performed at the KNK I reactor over a period of 18 month under different operational conditions. The signal fluctuations of the following transducers have been recorded: In-core and Ex-core neutron detectors, temperature-, flow-, pressure-, vibration- and acoustic sensors. These extensive measurements have been analyzed in the frequency range from 0,001 Hz to 1000 Hz with all currently known methods for the identification of noise sources. The following results have been found: - Neutron noise for $f < 1$ Hz is due to reactivity fluctuations only, for $f > 20$ Hz the white detection noise prevails. In the region from 1 Hz to 20 Hz the vibrations of core components contribute to neutron noise. - Neutron noise is influenced by the state of the plant. - The contributions to neutron noise due to the fluctuations of coolant flow and inlet temperature are small compared to those produced by the movements of the control rod initiated by the reactor control system. The quantitatively unidentifiable amount of reactivity fluctuations (0,6 time-dependent thermal bowing of the core. With respect to these results and by calculation of the neutron noise patterns to be expected for the SNR 300, the following possible applications for neutron noise analysis have been found: By means of neutron noise analysis only reactivity fluctuations can be identified and supervised which are produced by time dependent changes of the core geometry. Furthermore neutron noise analysis is well suited for a sensitive detection of control rod vibrations and of local sodium boiling. Finally it can be used for the surveillance of the proper functioning of the reactor control system and of the control rod drive mechanism. (orig./HP) 891 HP.

- 2198 (KFK--2619) POSSIBILITIES TO DETERMINE THE HEAT TRANSFER BETWEEN NUCLEAR FUEL ELEMENTS AND THE COOLANT IN A SODIUM COOLED REACTOR DURING NORMAL POWER OPERATION. Mitzel, F. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). Jun 1978. 45p. (In German). Dep. NTIS (US Sales Only), PC A04/MF A01.

In order to improve the safety, the surveillance and economy of sodium cooled reactors it is desirable to obtain a more precise knowledge of the heat transfer between the fuel and the coolant. Different possibilities have been investigated to determine the corresponding heat transfer coefficient by measuring the pertinent transfer functions. The result was the following: The heat transfer coefficient can be determined with reasonable accuracy at the KNK II and the SNR by means of small pseudostochastic power oscillations. These power oscillations must be artificially induced but do not require an interruption of the normal plant operation. The accuracy could however still be further improved by an additional measurement of the fuel temperature with a special instrumentation.

- 2199 (KFK--2660) INVESTIGATIONS OF DECAY HEAT REMOVAL BY NATURAL CONVECTION WITH BOILING IN SODIUM COOLED FAST REACTORS. Kaiser, A.; Peppier, W.; Strake, M. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). Mar 1979. 70p. (In German). Dep. NTIS (US Sales Only), PC A05/MF A01.

The safety analysis of a LMFR indicates the requirement of safely removing the decay heat produced after a reactor shut-down, especially in the case of a failure of all primary circuits. To investigate the conditions under which power in the range of the decay heat can be transferred from a pin bundle to a sodium loop by natural convection, a series of experiments was carried out. Special attention was paid to the behaviour of the natural convection system when boiling occurs, and also to the limits of cooling capability. To apply the experimental results a computer program was made using a simplified model of the emergency cooling system of the SNR 300. With this program several cases of emergency cooling under the boundary conditions of in-tank natural convection were analyzed, assuming a breach of a primary circuit. As an example, the consequences of an increase of the flow resistances in a subassembly were investigated. It was demonstrated that under conditions of steady state boiling there will be only very low vapour qualities. Similar results were obtained from investigations when the sodium temperature at the inlet to the core was elevated, and when the flow resistances in the cold leg of the natural convection loop were increased by a factor of two. Further experiments gave evidence that the cooling of the bundle will substantially be maintained under conditions of low vapour qualities. In summary, it may be stated that even under very pessimistic assumptions concerning the progress of the in-tank natural circulation, the cooling will be maintained reliably, even if boiling occurs for some time.

- 2200 (KFK--2662) THEORETICAL INVESTIGATIONS OF THE MELTOFF AND RESOLIDIFICATION PROCESS OF FUEL CLADDINGS DURING ACCIDENTS IN LIQUID METAL COOLED FAST BREEDER REACTORS. Angerer, G. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter; Karlsruhe Univ. (TH) (Germany, F.R.). Fakultät fuer Maschinenbau). Aug 1978. 182p. (In German). Dep. NTIS (US Sales Only), PC A09/MF A01.

Thesis.
During loss-of-coolant-flow accidents in

liquid metal cooled fast breeder reactors with failure to scram the fuel claddings will melt after boiling and evaporation of the coolant. The CMOT model presented here describes the subsequent process of relocation and resolidification of the molten claddings. The basic thermohydrodynamics equations of the two-phase flow of cladding material and sodium vapor are solved numerically by differential approximations in a Eulerian reference net. The results calculated by the model improved the insight into the dynamics of the cladding relocation process. Here are the main results: - Shortly after the onset of cladding relocation large waves of molten cladding material are generated. The motion of these waves contributes considerably to the material transport. - The dynamics of cladding relocation exhibits strong local incoherences. - The formation of cladding blockages observed at the ends of the fuel region is confirmed by the calculations. - In case of incoherent cladding meltoff less cladding material is transported upwards. - Cladding relocation strongly depends on the axial pressure drop and the underlying friction factor correlations. Recalculation of the R5 loss-of-coolant-flow experiment performed in the U.S. TREAT test reactor is in good agreement with the experimental data. (orig./HP) 891 HP.

- 2201 (KFK--2668) ANALYSIS OF SODIUM-VOID EXPERIMENTS IN ZPPR-3 MODIFIED PHASE 3 CORE. Yoshida, T. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Aug 1978. 45p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States--W. Germany Fast Reactor Exchange Program.

An analysis is presented of a series of sodium-void reactivity measurements performed in assembly 3 of Zero Power Plutonium Reactor (ZPPR-3), a mockup of the US Demoplant. In this series, large-zone sodium-void effects were studied in detail in the presence of many singularities, namely, control rods (CRs) and control rod positions (CRPs). The Karlsruhe data-and-method have been applied to an analysis of these experiments, and the results are presented. The work is aimed at complementing the sodium-void reactivity analysis based on the SNEAK experiments, where it was difficult to simulate a large plutonium-core of a prototype fast breeder reactor.

- 2202 (KFK--2684) FLUIDDYNAMIC INVESTIGATIONS WITH THE ABSORBERS OF THE FIRST AND SECOND SHUT DOWN UNIT OF THE SNR-300. Marten, K.; Hoffmann, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorbauelemente; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). Jan 1979. 26p. (In German). Dep. NTIS (US Sales Only), PC A03/MF A01.

Two independent shut down systems with different design characteristics will be installed in the SNR-300. The first shut down unit also controls the reactor. It consists of a rigid absorber and operates due to gravity from top downward into the core. The second shut down unit each is composed of three articulated absorbers installed beneath the core and pulled up into the core by acceleration springs. To support the design of the absorbers of the two shut down units the pressure losses, the mass flow distributions and the floating characteristics have been investigated experimentally in water flow. Beside the mass flow rates the main parameters represent the axial and radial positions of the

absorber bundles within the guide tube. The transferability of the results to sodium flow is guaranteed.

- 2203 (KFK--2690) ANNUAL REPORT OF THE WORKING GROUP 'FUEL PIN AND FUEL ELEMENT MECHANICS' OF THE INSTITUT FÜR REAKTORTECHNIK (IRT) OF THE TECHNISCHE HOCHSCHULE DARMSTADT FOR THE FAST BREEDER PROJECT. Fabian, H.; Humbach, W.; Lassmann, K.; Mueller, J.J.; Preusser, T.; Schmelz, K. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brüter). Sep 1978. 102p. (In German). Dep. NTIS (US Sales Only), PC A06/MF A01.

This report comprises six single lectures given at an information meeting organized by the Institut für Reaktortechnik der Technischen Hochschule Darmstadt (IRT) in Darmstadt on April 24, 1978. The lectures are an account of work performed at IRT on the mechanics of fuel pins and fuel elements and supported by the Fast Breeder Project (PSB) of KFK. These activities can be broken down into studies of the integral fuel pin (URANUS computer code) and into multidimensional studies of the fuel pin using the finite-element method (FINEL and ZIDRIG computer codes). Moreover, a report is presented of the status of the test facility for simulation of out-of-pile cladding tube loads and of the IRT project on the simulation and analysis of radiation damage.

- 2204 (KFK--2738) TP1, A COMPUTER PROGRAM FOR THE CALCULATION OF REACTIVITY AND KINETIC PARAMETERS BY ONE-DIMENSIONAL NEUTRON TRANSPORT PERTURBATION THEORY. Kobayashi, K. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. für Neutronenphysik und Reaktortechnik). Mar 1979. 76p. Dep. NTIS (US Sales Only), PC A05/MF A01.

Work performed under United States-West Germany Fast Reactor Exchange Program.

TP1, a FORTRAN-IV program based on transport theory, has been developed to determine reactivity effects and kinetic parameters such as effective delayed neutron fractions and mean generation time by applying the usual perturbation formalism for one-dimensional geometry. Direct and adjoint angular dependent neutron fluxes are read from an interface file prepared by using the one-dimensional S/sub n/-code DTK which provides options for slab, cylindrical and spherical geometry. Multigroup cross sections which are equivalent to those of the DTK-calculations are supplied in the SIGMN-block which is also read from an interface file. This block which is usually produced by the code GRUCAL should contain the necessary delayed neutron data, which can be added to the original SIGMN-block by using the code SIGMUT.

- 2205 (KFK--2762) DEVELOPMENT OF FILTERING DEVICES FOR CHEMICAL AGGRESSIVE AEROSOLS. Jordan, S.; Lindner, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Lab. für Aerosolphysik und Filtertechnik 1). Mar 1979. 25p. (In German). Dep. NTIS (US Sales Only), PC A03/MF A01.

Different types of filtering devices were tested to deposit chemical aggressive aerosols. Sodium fire aerosols were used as test aerosols. Glas fiber filters, wet scrubbers and sandbed filters were investigated. Commercially available fiber filters and scrubbers were used. Small sized multilayer sandbed filters were developed and optimized with respect to high deposition rate and load capacity and low pressure drop. The experimental results are

explained by theoretical calculations. The results of the three different precipitators are compared. Multilayer sandbed filters are with respect to deposition rate and pressure drop similar to fiber filters. Nevertheless the load capacity of a sandbed filter is much higher.

- 2206 (KFK--2773) CREEP AND STRESS-RUPTURE BEHAVIOR UNDER INTERNAL PRESSURE OF TUBES MADE FROM AUSTENITIC STAINLESS STEEL X8 CR NI MO NB 1616 (MATERIAL NO. 1.4981). Schaefer, L.; Polifka, F.; Kempe, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. für Material- und Festkörperforschung). May 1979. 88p. (In German). Dep. NTIS (US Sales Only), PC A05/MF A01.

Work performed under United States-EURFNR Fast Breeder Reactor Exchange Program.

Creep and stress rupture tests have been performed at 600, 650, 700 and 750°C on tubes made from three different heats from the austenitic stainless steel X8 Cr Ni Mo Nb 1616 (Material No. 1.4981). The tubes were loaded by internal pressure and the tangential (hoop) creep strain was measured continuously. The results are presented in form of creep curves, stress-time to rupture curves and curves for a creep limit. The average and minimum creep rates as a function of the applied stress have been evaluated and are described with a creep law analogous to Norton's creep law. An interpolation and extrapolation of the stress-rupture-strength and the creep strength are possible using the time-temperature-parameter-plot after Larson and Miller.

- 2207 (KFK--2794) CRISS POWER SPECTRAL DENSITY: A METHOD FOR CORRELATING SIGNAL COMPONENTS FROM DIFFERENT FREQUENCY RANGES. Vaeth, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. für Neutronenphysik und Reaktortechnik). Apr 1979. 16p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States-West Germany Fast Reactor Exchange Program.

A special method is described for measuring the correlation of two signal components at different frequencies: the CRISS power spectral density. From this new function in frequency analysis, the correlation of two components can be determined quantitatively either they stem from one signal or from two diverse signals. The principle of the method, suitable for the higher harmonics of a signal as well as for any other frequency combination is shown for the digital frequency analysis technique. Two examples of CRISS power spectral densities demonstrate the operation of the new method.

- 2208 (KFK--2802-B) PUBLICATION LIST CONCERNING LMFBR-WORK OF KFK FROM 1976 UNTIL 1978. Hueper, R. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brüter). Apr 1979. 11p. (In German). Dep. NTIS (US Sales Only), PC A10/MF A01.

This list is a continuation of the lists of the Karlsruhe Fast Breeder Project publications given in the earlier reports KFK-Ext. 25/7-1 and 25/76-1. Corresponding to the present organization of the project in the framework of the cooperation between Germany, Belgium, the Netherlands and France, the list is subdivided into the fields of the working groups of the 'Arbeitsausschuss F+E-Programme des SNR-300'. In the last part of the list general surveys of the project work and its objectives are quoted.

- 2209 (KFK--2817) GERMAN OXIDE FUEL PIN IRRADIATION TEST EXPERIENCE FOR FAST REACTORS. Kummerer, K. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). Mar 1979. 27p. Dep. NTIS (US Sales Only), PC A02/MF A01. Work performed under United States-UKAEA Fast Reactor Exchange Program.

After a short introduction concerning features and problems in oxide fuel pin design, the experiments are described in some detail, including the design of the experimental pins, the parameter variations within the test groups, the irradiation conditions, and also some characteristic results. In order to get an easy survey, a lot of tables combined with schematic design drawings are presented. After that a nomographical synopsis indicates the regions of parameters covered. Finally conclusions are drawn concerning the experimental basis for a sound reference oxide fuel pin design. Out of the body of present knowledge the future trends and needs in developmental effort are indicated.

- 2210 (KFK--2820) OBSERVATIONS OF THE BEHAVIOR OF GAS IN THE WAKE BEHIND A CORNER BLOCKAGE IN FAST BREEDER SUBASSEMBLY GEOMETRY. Fukuzawa, Y. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). Jul 1979. 44p. NTIS (US Sales Only), PC A03/MF A01.

Observations were made of gas behavior in the wake behind a 21% corner blockage in the subassembly geometry of a liquid metal fast breeder reactor. The test section used represented one half of the reactor fuel subassembly, divided along the vertical plane of symmetry through the blockage. A glass wall occupied the position of this plane. Water was allowed to flow between glass rods simulating fuel pins, the velocity being changed from 1.2 to 4.5 m/s. Argon was injected into the wake or into the flow upstream of the blockage, the injection rate being changed from 1 to 230 Ncm³/s (standard temperature and pressure).

- 2211 (KFK--2824) DEVELOPMENT OF A CODE FOR DESCRIPTION OF SODIUM SPRAY AND POOL FIRES. PART I: COMPARISON BETWEEN THE CODES SOFIRE II AND NABRAND. Alexas, A. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Lab. fuer Aerosolphysik und Filtertechnik 1). Sep 1979. 30p. (In German). Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States-EURFN Fast Breeder Reactor Exchange Program.

In the scope of the development of a code to describe both, sodium pool- and spray fires, the ell-known codes SOFIRE II and NABRAND have been compared. Regarding the program technique of both codes, the NABRAND-code seems to be the better one, though it includes some conservatism in the modeling and in the transport coefficients used. For a realistic estimation of the consequences of large sodium fires in an LMFBR, an elimination of these conservatism is necessary. After that it must be investigated if a combination of the modified version of the NABRAND-code and of a spray fire-code (for example the code SPRAY) is efficient.

- 2212 (KFK--2827) DEPENDENCE OF DIFFUSION THEORY RESULTS ON THE MESH SIZE FOR FAST REACTOR CALCULATIONS. Buckel, G.; Gebhardt, W.; Kiefhaber, E. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Jul 1979. 49p. NTIS (US Sales Only), PC A03/MF

A01.

In investigations concerning sodium void reactivities for SNEAK-9C2-assemblies inconsistencies were detected in the results coming from Δk of successive diffusion and exact perturbation calculations, respectively. Therefore, discretization and rounding errors in neutronic reactor calculations and their effects on numerical results are considered for a well known SNR-300 type benchmark problem as well as for the slightly simplified original problem.

- 2213 (KFK--2834) EXPERIMENTAL INVESTIGATIONS OF TURBULENT FLOWS IN ROD BUNDLES WITH AND WITHOUT SPACER GRIDS. Trippe, G. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorbauelemente; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter; Karlsruhe Univ. (TH) (Germany, F.R.). Fakultae fuer Maschinenbau). Jul 1979. 92p. (In German). Dep. NTIS (US Sales Only), PC A06/MF A01.

Thesis.

In the thermofluiddynamic design of liquid metal cooled reactor fuel elements the lack of experimentally confirmed knowledge of the three-dimensional flow events in rod bundles provided with spacer grids has appeared as a significant problem. To close this gap of knowledge, detailed measurements of the local velocities were made on a 19-rod bundle model. The Pitot method of differential pressure measurements was used as the measuring system. In the first part of the work the fully developed flow regime not influenced by spacers was investigated. A simple relation was derived for distributing the mass flow among the subchannels of a rod bundle; it is but slightly dependent on the Reynolds number. This relation allows a quick, coarse calculation of the distribution of the undisturbed, fully developed mass flow in bundles with similar geometries. By evaluation of further experiments known from the literature, empirical relationships were found for the local velocity distribution within the subchannels of such bundles. In the second part the effect of grid shaped spacers was investigated. The three-dimensional flow events caused by the spacers were completely recorded and interpreted physically. The deeper understanding of these flow processes can now serve to improve the model concept used in the present design computer programs. Single results of the investigations which take primary importance are the quantitative relations existing between the changes of mass flow in the bundle boundary zone, caused by a spacer, and the geometry of this spacer. The transferability to other bundle geometries was discussed and delimited. Moreover, it was shown that the mass flow in the bundle boundary zone can be successively reduced by spacers placed one behind the other in the bundle. A noticeable dependence of flow events on the Reynolds number was not found for the range relevant in practical application ($30.000 \leq Re \leq 60.000$).

- 2214 (KFK--2834) EXPERIMENTAL INVESTIGATIONS OF TURBULENT FLOWS IN ROD BUNDLES WITH AND WITHOUT SPACER GRIDS. Trippe, G. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorbauelemente). Jul 1979. 105p. (In German). Dep. NTIS (US Sales Only), PC A06/MF A01.

In the thermofluiddynamic design of liquid metal cooled reactor fuel elements the lack of experimentally confirmed knowledge of the three-

dimensional flow events in rod bundles provided with spacer grids has appeared as a significant problem. To close this gap of knowledge, detailed measurements of the local velocities were made on a 19-rod bundle model. The Pitot method of differential pressure measurements was used as the measuring system.

- 2215 (KFK--2841) HEAT TRANSFER FROM INTERNALLY HEATED FLUIDS WITH TEMPERATURE DEPENDENT VISCOSITY. Fieg, G. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Aug 1979. 23p. (In German). Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States-EURFN Fast Breeder Reactor Exchange Program.

Thermo-hydraulic experiments on heat transfer in fluids with high Prandtl-numbers and strong temperature dependent viscosity are described. Pure Benard-convection and volume heated liquids have been investigated. In both cases the results show no essential differences compared to the heat transfer in fluids with nearly temperature independent viscosity.

- 2216 (KFK--2858) CONTRIBUTION TO THE STRUCTURAL ANALYSIS OF CYLINDRICAL FUEL RODS OF NUCLEAR REACTORS WITH CONSIDERATION OF MULTIDIMENSIONAL BEHAVIOR, BASED ON FINITE-ELEMENT-METHOD. Fabian, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). Sep 1979. 208p. (In German). Dep. NTIS (US Sales Only), PC A10/MF A01.

Work performed under United States-EURFN Fast Reactor Breeder Reactor Exchange Program.

A critical review of the state-of-the-art in fuel rod structural analysis shows that in general analyses are based on one-dimensional models. This investigation considers the two-dimensional aspects of the fuel rod design. Therefore the FEM-codes, ZIDRIG and FINEL, have been developed. For the cladding the influences of manufacturing tolerances, integral ribs and end-plugs as well as the behavior during LOCA (ballooning) have been calculated. The pellets have been analyzed with their real dimensions and with consideration of plasticity and cracks. Finally the mechanical pellet-cladding-interaction is considered, i.e., bamboo like deformation.

- 2217 (KFK-EXT--8/76-4) LMFBR POST ACCIDENT HEAT REMOVAL TESTING NEEDS AND CONCEPTUAL DESIGN OF A TEST FACILITY. Kleefeldt, K.; Kuechle, M.; Royle, P.; Werle, H.; Boenisch, G.; Heinzel, V.; Mueller, R.A.; Schramm, K.; Smidt, D. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). Mar 1977. 196p. Dep. NTIS (US Sales Only), PC A09/MF A01.

A study has been carried out in which the needs and requirements for a test facility were derived, enabling detailed investigation of key phenomena anticipated during the post accident heat removal (PAHR) phase as a consequence of a postulated LMFBR whole core accident. Part I of the study concentrates on demonstrating the PAHR phenomena and related testing needs. Three types of experiments were identified which require in-pile testing, ranging from 10 to 70 cm test bed diameter and correspondingly, 30 to 5 W/g minimum power density in the test fuel. In part II a conceptual design for a test facility is presented, emphasizing the capability for accomodating large test beds. This is achieved by a below-reactor-vessel

testing device, neutronically coupled to a 100 Mwt sodium cooled fast reactor.

- 2218 (KFK-EXT--8/78-4) MEASUREMENT OF HEAT FLUX DISTRIBUTION OF A PROTOTYPE HEATER ROD WITH SINUSOIDAL AXIAL POWER PROFILE. Gonzalez, S. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). Oct 1978. 29p. Dep. NTIS (US Sales Only), PC A03/MF A01.

To support the safety analysis of the SNR-300 prototype sodium cooled fast breeder reactor core, experiments concerning integral boiling events will be performed in a 37 electrically heated rod bundle. For this purpose heater rods having a sinusoidal axial heat flux distribution have been developed in accordance with the SNR-300 fuel pins. The variation in the heat flux is attained by varying the band width of the conductor element. The heat flux distribution of one already manufactured heater rod has been measured and compared with an equation which describes the required distribution. It was found that the differences between the measurements and the theoretical values in the range of 50 to 850 mm of the heated length were less than 7%. It is concluded that the heater rod can simulate satisfactorily the SNR-300 fuel pin.

- 2219 (KFK-EXT--8/78-5) ASSESSMENT OF STRUCTURAL DYNAMICS PROBLEMS IN NUCLEAR REACTOR SAFETY. Liebe, R. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung; Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). Oct 1978. 56p. (In German). Dep. NTIS (US Sales Only), PC A04/MF A01.

The paper discusses important physical features of structural dynamics problems in reactor safety. First a general characterization is given of the following problems: Containment deformation due to pool-dynamics during BWR-blowdown; behavior of the core internals due to PWR-blowdown loads; dynamic response of a nuclear power plant during an earthquake; fuel element deformation due to local pressure pulses in an LMFBR core. Several criterias are formulated to classify typical problems so that a better choice can be made both of appropriate mathematical/numerical as well as experimental techniques. The degree of physical coupling between structural dynamics and fluid dynamics is discussed in more detail since it requires particular attention when selecting problem-oriented methods of solution. Some examples are given to illustrate the application and to compare advantages and disadvantages of several numerical methods. Then description is given of experimental techniques in structural dynamics and typical problem areas are identified. Finally some results are presented concerning the fuel element deformation problem in LMFBRs and from the general considerations some important conclusions are summarized. (orig.) 89i RW 892 AP.

- 2220 (KFK-tr--521) RATE OF REACTIVITY CHANGE OF THE BOR-60 REACTOR DURING THE OPERATION CYCLE. Afanas'ev, V.A.; Efimov, V.N.; Krasnoyarov, N.V.; Skulkin, N.N.; Fedyakin, R.E. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Internationales Buero/Uebersetzergruppe). 1977. 6p. (In German). Dep. NTIS (US Sales Only), PC A02/MF A01.

The result of measurements and calculations of the reactivity reserve at the BOR-60 reactor is reported. The results of the computations of the rate of change of the isotopic composition of the fuel during burn-up depending on the inventory of fissile material and the results of the computations aimed at showing the influence of radiation-induced volumetric swelling of the fuel on the reactivity change are represented graphically. The results of measurements and computations are compared with one another.

- 2221 (KFK-tr--575) RADIOACTIVE IMPURITIES ACCUMULATION IN BOR-60 COLD TRAP. Kizin, V.D.; Lisitsyn, E.S.; Poznyak, G.I.; Polyakov, V.I.; Chechetkin, Y.V. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Internationales Buero/Uebersetzergruppe). Jun 1978. 28p. (In German). Dep. NTIS (US Sales Only), PC A03/MF A01.

Radioactivity accumulation and distribution in BOR-60 cold trap have been studied. Cesium, iodine and some other radionuclide decontamination factors obtained using a standard cold trap in sodium coolant have been determined. Laboratory and loop experiments have been carried out to explain processes of radioactivity accumulation in the trap.

- 2222 (KFK-tr--576) RADIOACTIVITY OF LONG-LIVED NUCLIDES IN THE PRIMARY CIRCUIT OF THE REACTOR BOR-60 DURING OPERATION WITH DEFECTIVE FUEL ELEMENTS. Gryazev, V.M.; Kizin, V.D.; Lisitsyn, E.S.; Polyakov, V.I.; Chechetkin, Y.V. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Internationales Buero/Uebersetzergruppe). Jun 1978. 21p. (In German). Dep. NTIS (US Sales Only), PC A02/MF A01.

The summarized results of measurements of the enrichment and distribution of radioactive nuclides from corrosion and of fission products during the four years of operation of BOR-60, including a longer period of operation with defective fuel elements in the core, are presented. It is shown that for operation with approximately 1% leaking fuel rods radiation exposure becomes worse mainly because of release and enrichment of cesium isotopes in the coolant. Of the other fission products, the largest contribution to the dose rate in pipework and components is given by ^{140}Ba / ^{140}La and ^{95}Nb . On operation with 0.1 to 0.2% of leaking fuel rods, this contribution is comparable to that of the corrosion products ^{60}Co and ^{54}Mn . The radioactivity of corrosion products in the circuit has not increased within the last three years and was about one order of magnitude lower than the theoretical values. The corrosion and fission products are nonuniformly distributed over the circuit. Concentration of ^{95}Nb and ^{60}Co in the pipe for 'cold' sodium is larger by a factor of 2 - 5 and of ^{140}Ba and ^{54}Mn by a factor of 10-20 than in the pipes for 'hot' sodium. Most of the cobalt was found to deposit in the heat exchanges. The effectiveness of emptying the pipes from coolant in order to reduce the dose rate is assessed.

- 2223 (KFK-tr--578) SOME PROBLEMS OF OPERATION BEHAVIOUR OF HEXAGONAL BOXES FOR FUEL ELEMENTS OF FAST POWER REACTORS. Proshkin, A.A.; Zabud'ko, L.M. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Internationales Buero/Uebersetzergruppe). Jun 1978. 59p. (In German). Dep. NTIS (US Sales Only), PC A04/MF A01.

The present work shows a method to determine the operational behaviour of hexagonal fuel

cans for fast reactors by using tensions caused by the coolant pressure, and tensions due to a possible interaction of the fuel rod bundle and the can with axial symmetrical temperature distribution, taking into account the change of physical and mechanical, material properties of the can during irradiation. A few questions on the statistical assessment of the operational behaviour of the can are considered, based on which one can determine the probability of damaging the fuel cans during reactor operation and the contribution of various factors on the non-safeness in calculating the cans' operational behaviour. An example is given for the calculation of a fuel can for a large power plant based on the method shown.

- 2224 (LA--4319) URANIUM DIOXIDE EQUATION OF STATE FOR LMFBR EXCURSION ANALYSIS. Goplen, B. (Los Alamos Scientific Lab., NM (USA)). Jul 1969. Contract W-7405-ENG-36. 19p. Dep. NTIS, PC A02/MF A01.

Declassified 10 Jul 1974.

Reactor excursion analysis, which is essentially a coupled neutronics-hydrodynamics problem, is highly dependent upon the assumed equation of state for the reactor fuel. Unfortunately, equation-of-state information is not experimentally available for most reactor fuels. However, by using dynamic shock-wave techniques, it has been possible to measure the Hugoniot equation of state of uranium dioxide. Extension of the Hugoniot data along pressure release loci then allows an estimation of the equation of state in the desired high-temperature, low-pressure regime.

- 2225 (LA--6924-PR) EXAMINATION OF FAST REACTOR FUELS, FBR ANALYTICAL QUALITY ASSURANCE STANDARDS AND METHODS, AND ANALYTICAL METHODS DEVELOPMENT: IRRADIATION TESTS. PROGRESS REPORT, APRIL 1--JUNE 30, 1977. Baker, R.D. (comp.). (Los Alamos Scientific Lab., NM (USA)). Aug 1977. Contract W-7405-ENG-36. 14p. Dep. NTIS, PC A02/MF A01.

Characterization of unirradiated and irradiated LMFBR fuels by analytical chemistry methods will continue, and additional methods will be modified and mechanized for hot cell application. Macro- and microexaminations will be made on fuel and cladding, using the shielded electron microprobe, emission spectrograph, radiochemistry, gamma scanner, mass spectrometers, and other analytical facilities. New capabilities will be developed in gamma scanning, analyses to assess spatial distributions of fuel and fission products, mass spectrometric measurements of burnup and fission gas constituents and other chemical analyses. Microstructural analyses of unirradiated and irradiated materials will continue, using optical and electron microscopy and autoradiographic and x-ray techniques. Analytical quality assurance standards are designed to assure the quality of the chemical characterizations necessary to evaluate reactor components relative to specifications. Tasks include: (1) the preparation and distribution of calibration materials and quality control samples for use in quality assurance surveillance programs, (2) the development of and the guidance in the use of quality assurance programs for sampling and analysis, (3) the development of improved methods of analysis, (4) the preparation of continuously updated analytical method manuals, and (5) maintenance of referee analysis capabilities to characterize on a reimbursable basis, materials in dispute between fabricators (vendors) and reactor operators (purchasers). Highly reliable analytical methods for the measurement of burnup, various gases, and trace impurities in

irradiated fuels are also being developed.

- 2226 (LA--7038-PR) EXAMINATION OF FAST REACTOR FUELS, FBR ANALYTICAL QUALITY ASSURANCE STANDARDS AND METHODS, AND ANALYTICAL METHODS DEVELOPMENT: IRRADIATION TESTS. PROGRESS REPORT, JULY 1--SEPTEMBER 30, 1977. Baker, R.D. (comp.). (Los Alamos Scientific Lab., NM (USA)). Nov 1977. Contract W-7405-ENG-36. 22p. Dep. NTIS, PC A02/MF A01.

Information is presented concerning the examination of fast reactor fuels, FBR analytical quality assurance standards, and analytical methods development irradiation tests.

- 2227 (LA--7133-MS) FEASIBILITY STUDY FOR A POSTACCIDENT HEAT REMOVAL FACILITY. Barts, E.W.; Apperson, C.E. Jr.; Dunwoody, W.E.; Bennett, J.G. (Los Alamos Scientific Lab., NM (USA)). Jan 1978. Contract W-7405-ENG-36. 44p. Dep. NTIS, PC A03/MF A01.

An initial feasibility investigation for PAHRTEF, a Postaccident Heat Removal Test Facility, is presented. The facility would provide an experimental capability for PAHR experiments beyond that available in any currently existing or proposed U.S. safety test facility. The facility design presented in this report is based upon the technology developed for the ROVER nuclear rocket propulsion program. The core is a graphite-moderated, helium-cooled, epithermal core with radial reflector control. The PAHR experiments are located just below the reactor containment vessel, very near the bottom of the core. The experiments (up to 55% enriched) are driven and controlled by neutrons leaking axially from the core such that the PAHRTEF core and the experiment form a coupled reactor system. The experiment can be designed so that it is extremely unlikely that the test fuel by itself could form a critical system. The investigation indicates that adequate fission heating of large PAHR experiments could be provided at low driver core power levels. Both the reactor and the experiment handling and examination equipment can use available technology and, whenever possible, existing equipment and buildings.

- 2228 (LA--7161-MS) LOS ALAMOS FAILURE MODEL (LAFM): A CODE FOR THE PREDICTION OF LMFBR FUEL PIN FAILURE. Mast, P.K. (Los Alamos Scientific Lab., NM (USA)). Feb 1978. Contract W-7405-ENG-36. 47p. Dep. NTIS, PC A03/MF A01.

A new fuel pin failure model (the Los Alamos Failure Model) has been developed, a preliminary version of which is reported in this document. The model treats pin failure due to fuel-cladding differential thermal expansion, fission-gas pressurization, and cladding thermal stress. Radial fuel cracking and crack closure, as well as the presence of an initial circumferential fuel crack, are accounted for. The elastic-plastic stress-strain behavior of the cladding is calculated with a single node representation of the cladding using the Tresca Yield criterion. This results in a very fast running computer code with small storage requirements.

- 2229 (LA--7171-PR) EXAMINATION OF FAST REACTOR FUELS, FBR ANALYTICAL QUALITY ASSURANCE STANDARDS AND METHODS, AND ANALYTICAL METHODS DEVELOPMENT: IRRADIATION TESTS. PROGRESS REPORT 1 OCT--31 DEC 1977. Baker, R.D. (Los Alamos Scientific Lab., NM (USA)). Mar 1978. Contract W-7405-ENG-36. 19p. Dep. NTIS, PC

A02/MF A01.

Progress is summarized in the following areas: (1) examination of fast reactor fuels, (2) FBR analytical quality assurance standards and methods, and (3) analytical methods and irradiation testing techniques. (DG)

- 2230 (LA--7194-MS) APPLICATION OF THE SIMMER-I CODE TO LOF ACCIDENTS IN AN LMFBR. Henninger, R.J.; Alcouffe, R.E. (Los Alamos Scientific Lab., NM (USA)). Feb 1978. Contract W-7405-ENG-36. 22p. Dep. NTIS, PC A02/MF A01.

In the area of mechanical disassembly phase energetics, a comprehensive comparison and test against the VENUS-II code have been completed. The conclusions are that for energetic excursion (70s/s), the two calculations give essentially the same results. Comparisons to an isentropic expansion have also been made for the mechanical energy conversion as a result of expanding fuel vapor. SIMMER induced mixing effects are dominant when a temperature distribution is assumed for an expansion initial condition. The SIMMER code can be made to compare well with an isentropic expansion assuming a spatially uniform temperature. In the predisassembly models task, the SIMMER mockup of a SLUMPY calculation of initial fuel motion has been carried out. The results of the two calculations do not compare well because of incompatibilities in the SIMMER-I and SLUMPY models.

- 2231 (LA--7266-PR) EXAMINATION OF FAST REACTOR FUELS, FBR ANALYTICAL QUALITY ASSURANCE STANDARDS AND METHODS, AND ANALYTICAL METHODS DEVELOPMENT: IRRADIATION TESTS. PROGRESS REPORT, JANUARY 1--MARCH 31, 1978. Baker, R.D. (comp.). (Los Alamos Scientific Lab., NM (USA)). May 1978. Contract W-7405-ENG-36. 14p. Dep. NTIS, PC A02/MF A01.

Progress is reported for the following tasks: equipment development, analytical chemistry program for mixed oxide fuel, analytical chemistry program for boron carbide, determination of burnup, and development of measurement techniques for fission gas release.

- 2232 (LA--7564-PR) TRANSPORT AND REACTOR THEORY. PROGRESS REPORT, JULY 1--SEPTEMBER 30, 1978. Miller, W.F. Jr.; Dudziak, D.J.; O'Dell, R.D.; Alcouffe, R.E.; Gomez, M.F. (Los Alamos Scientific Lab., NM (USA)). Nov 1978. Contract W-7405-ENG-36. 37p. Dep. NTIS, PC A03/MF A01.

Progress is summarized in the areas of fission reactor neutronics, radiation shielding, fusion reactor nucleonics, reactor safety, and supporting research. (DG)

- 2233 (LA--7569-PR) EXAMINATION OF FAST REACTOR FUELS, FBR ANALYTICAL QUALITY ASSURANCE STANDARDS AND METHODS, AND ANALYTICAL METHODS DEVELOPMENT, IRRADIATION TESTS. PROGRESS REPORT, JULY 1--SEPTEMBER 30, 1978. Baker, R.D. (comp.). (Los Alamos Scientific Lab., NM (USA)). Nov 1978. Contract W-7405-ENG-36. 19p. Dep. NTIS, PC A02/MF A01.

Since the first quarterly report (LA--5798-PR), the Analytical Quality Assurance Standards and Methods project has been divided into the following two projects: FBR Analytical Quality Assurance Standards and Methods; and Analytical Methods Development, Irradiation Tests. Most of the investigations discussed here are of the continuing type. Results and conclusions described may therefore be changed or augmented as the work continues.

- 2234 (LA--7570-PR) ADVANCED FAST REACTOR FUELS PROGRAM. SECOND ANNUAL PROGRESS REPORT, JULY 1, 1975--SEPTEMBER 30, 1976. Baker, R.D. (comp.). (Los Alamos Scientific Lab., NM (USA)). Dec 1978. Contract W-7405-ENG-36. 114p. Dep. NTIS, PC A06/MF A01.

Results of steady-state (EBR-II) irradiation testing, off-normal irradiation design and testing, fuel-cladding compatibility, and chemical stability of uranium-plutonium carbide and nitride fuels are presented.

- 2235 (LA--7678-PR) EXAMINATION OF FAST REACTOR FUELS, FBR ANALYTICAL QUALITY ASSURANCE STANDARDS AND METHODS, AND ANALYTICAL METHODS DEVELOPMENT: IRRADIATION TESTS. PROGRESS REPORT, OCTOBER 1--DECEMBER 31, 1978. Baker, R.D. (comp.). (Los Alamos Scientific Lab., NM (USA)). 1978. Contract W-7405-ENG-36. 11p. Dep. NTIS, PC A02/MF A01.

This is the seventeenth quarterly report on the Examination of Fast Reactor Fuels and Fast Breeder Reactor (FBR) Analytical Quality Assurance and Methods of the Los Alamos Scientific Laboratory. Since the first quarterly report (LA-5798-PR), the Analytical Quality Assurance Standards and Methods project has been divided into the following two projects: FBR Analytical Quality Assurance Standards and Methods; and Analytical Methods Development - Irradiation Tests. Most of the investigations discussed here are of the continuing type. Results and conclusions described may therefore be changed or augmented as the work continues.

- 2236 (LA--7827-PR) EXAMINATION OF FAST REACTOR FUELS AND FBR ANALYTICAL QUALITY ASSURANCE STANDARDS AND METHODS. PROGRESS REPORT, JANUARY 1--MARCH 31, 1979. Baker, R.D. (comp.). (Los Alamos Scientific Lab., NM (USA)). May 1979. Contract W-7405-ENG-36. 15p. Dep. NTIS, PC A02/MF A01.

This project is directed toward the examination and comparison of the effects of neutron irradiation on Liquid Metal Fast Breeder Reactor (LMFBR) Program fuel materials. Unirradiated and irradiated materials will be examined as requested by the Reference Fuels System Branch of the Division of Reactor Research and Technology (DRRT). Reliable chemical characterizations are essential to the development of high-quality fuels, control rods, and other reactor components in the FBR Program. The objective tasks are designed to assure the quality of the chemical characterizations necessary to evaluate reactor components relative to specifications.

- 2237 (LA--7861-MS) SPECTRA-IV, BENCHMARK SPECTRA BASED ON LIB-IV. Kidman, R.B. (Los Alamos Scientific Lab., NM (USA)). Jun 1979. Contract W-7405-ENG-36. 99p. Dep. NTIS, PC A05/MF A01.

Flux and adjoint spectra for each of the Cross Section Evaluation Working Group fast reactor benchmarks are numerically and graphically presented in this handbook. The spectra were computed using a 50-group cross-section library (LIB-IV) that was generated with the MINX processing code using ENDF/B-IV data. The graphical figures compare smooth interpolated curves for the central flux, edge flux, central adjoint, transport reaction rate, and original cross-section averaging weighting function.

- 2238 (LA--7901-MS) PRELIMINARY FABRICATION STUDIES OF ALTERNATIVE LMFBR CARBIDE FUELS.

Gutierrez, R.L.; Herbst, R.J.; Johnson, K.W.R. (Los Alamos Scientific Lab., NM (USA)). Oct 1979. Contract W-7405-ENG-36. 11p. Dep. NTIS, PC A02/MF A01.

Preliminary fabrication studies were made of various compositions of thorium-uranium carbide and thorium-plutonium carbide fuel pellets that were prepared using the carbothermic reduction process. Temperatures of 1750 and 2000°C were used during the reduction cycle. Sintering temperatures of 1800 and 2000°C were used to prepare fuel pellets of low (87%) and high (> 94%) theoretical densities.

- 2239 (LA--7938-C) SPECIALISTS' WORKSHOP ON PREDICTIVE ANALYSIS OF MATERIAL DYNAMICS IN LMFBR SAFETY EXPERIMENTS, MARCH 13-15, 1979. Stevenson, M.G. (comp.). (Los Alamos Scientific Lab., NM (USA)). Jul 1979. Contract W-7405-ENG-36. 343p. Dep. NTIS, PC A15/MF A01.

Separate abstracts are included for each of the papers presented concerning the performance of reactor components and materials under accident conditions.

- 2240 (LA--7938-C, pp 9-20) FUEL CRACK VOLUME PREDICTIONS AND MEASUREMENTS. Fox, G.L.; Randklev, E.H. (Hanford Engineering Development Lab., Richland, WA). Jul 1979. Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

Void volume within an FBR fuel pin has a substantial effect on calculations of cladding strain and lifetime for transient overpower accident analyses. A substantial fraction of this void volume is contained within startup cracks that are still present during fuel irradiation. This paper discusses a crack prediction method, crack measurement techniques, compares predictions with measurements, and includes a sufficient crack fraction data base to be of use to other fuel pin behavior analysts.

- 2241 (LA--7938-C, pp 21-38) INFLUENCE OF THERMAL CRACKING ON TRANSIENT PRE-FAILURE MOLTEN FUEL MOTION. Kuczera, B.; Billaux, M. (Kernforschungszentrum, Karlsruhe, Germany). Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

Under hypothetical transient conditions analyzed in the frame of safety investigations of LMFBRs, the fuel pin will undergo various modifications which influence remarkably the instant and the location of the clad failure. The essential part of the present contribution concerns cracked fuel behaviour and transient pre-failure fuel relocation. Enhanced by experimental evidences a new transient model, called BREDA-III A, was developed which allows a more detailed simulation of the fuel pin behaviour under accident conditions until pin failure. The application of the model has led to a better understanding of the sequence of physical events observed during moderate transient overpower (TOP) experiments.

- 2242 (LA--7938-C, pp 39-51) ANALYSIS OF INTERNAL FUEL MOTION DURING PINEX-2 EXPERIMENT. Padilla, A. Jr.; Baars, R.E.; Porten, D.R.; Randklev, E.H. Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

This paper describes the analyses performed for the PINEX-2 experiment to calculate the

ejection of molten fuel into the reflector and fission gas plenum for an internally-vented fuel pin during a simulated 5\$/s transient overpower excursion. The LAFM code was used to predict the transient fuel melting and fission gas release, and the HOTPIN and FUMO-T codes were used to predict the fuel ejection. The analytical results were compared with initial data from both the Pinhole-TV Imaging System and the fast-neutron hodoscope, as well as post-transient examinations of the fuel pin.

- 2243 (LA--7938-C, pp 52-66) PINEX II: FRENCH CONTRIBUTION TO AN INTERNATIONAL BENCHMARK EXPERIMENT. Melis, J.C.; Meyer-Heine, A. Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

This note presents briefly the code PHYSURA used to interpret safety experiments. As an application, the French contribution to an international benchmark test, PINEX 2, is described. The results show that a simple code like PHYSURA gives good results even for such a complex experiment.

- 2244 (LA--7938-C, pp 67-84) PINEX-AR, A FUEL DYNAMICS CODE APPLIED TO PINEX-2 PREDICTION AND REACTOR ANALYSIS. Coddington, P. Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

The PINEX-AR code is a fuel dynamics code that incorporates a heat transfer calculation to provide fuel and clad temperatures, a fission gas release routine, a method for determining the gas pressure within the molten fuel, the evaluation of the movement of molten fuel along the fuel pin and breeder central hole, and a determination of clad failure by reference to clad strain and molten fuel melt through. The code has been successfully applied to the PINEX-2 transient and demonstrates that molten fuel will be pushed into the gas plenum region and that the fuel pin clad is not expected to fail. The code has also been applied to fast reactor overpower transients and shows that molten fuel behavior similar to PINEX-2 can be expected for fast transients, whereas for slow transients the central hole is predicted to plug due to frozen fuel.

- 2245 (LA--7938-C, pp 85-98) CLAD STRAIN AND MELT-THROUGH FAILURE MODE ANALYSIS FOR FAST-RUNNING APPLICATION. Moorhead, T.P. (Atomic Energy Establishment, Winfrith, England). Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

A system of calculations representing various phenomena affecting fuel pin failure in accident transients is described, which is simple and fast enough for repetitive calls and yet retains a sound physical basis with experimental background. Its current implementation in the whole-core accident code FRAX-2 is illustrated.

- 2246 (LA--7938-C, pp 99-115) FRESS PIN FAILURE MODEL AND ITS APPLICATION TO E-8 TREAT TEST. Kalimullah. (Argonne National Lab., IL). Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

FRESS is a cladding rupture prediction model for an irradiated mixed-oxide LMFBR fuel pin

during transient heating based only on the internal pressurization of the cladding by the fission gas released from fuel grains during the transient. The model is applied to the analysis of the hottest PNL-10-53 pin in the 7-pin E-8 TREAT test which simulates a 3\$/sec transient overpower. Although the uncertainties of the inputs to the temperature calculation done with the COBRA code have not been included, the uncertain input parameters to FRESS have been varied over their estimated uncertainties. The cladding rupture predictions are a few tens of milliseconds late compared to the most probable failure time detected in the test. However, these calculations seem to indicate that fission gas pressure is a significant mechanism for causing clad rupture in this test.

- 2247 (LA--7938-C, pp 116-121) FUEL PIN MODELING OF THE FRESH OXIDE/SODIUM PROMPT BURST ENERGETICS EXPERIMENTS. Young, M.F. (Sandia Labs., Albuquerque, NM). Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

A series of twelve Prompt Burst Energetics (PBE) experiments utilizing fresh uranium dioxide fuel in stagnant sodium coolant have been performed in Sandia Laboratories' Annular Core Pulse Reactor (ACPR). PBE experiments are designed to study the energetic response of important reactor fuel-clad-coolant systems under superprompt critical conditions. The direct aim is to identify and characterize the phenomena resulting in pressure generation and the conversion of thermal energy to work. Other PBE experiments have involved fresh UO_2 fuel in a helium filled capsule and fresh UC fuel in stagnant sodium. This paper presents results of this series of experiments and related analysis done with a pin model, EXPAND, developed to predict pin failure time and location for fresh fuel pins. The model also gives temperatures and pressures at failure for the fuel-clad-coolant system, which are used as initial conditions for analysis of the post-failure behavior of the ruptured fuel pin.

- 2248 (LA--7938-C, pp 122-124) SODIUM BOILING WITH STAINLESS STEEL CLAD MELTING AND RELOCATION OUT-OF-PILE EXPERIMENTAL RESULTS AND INTERPRETATION. Seiler, J.M. (Commissariat a l'Energie Atomique, Grenoble, France). Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

A feasibility test of out-of-pile sodium boiling experiments with stainless steel clad melting has been successfully achieved at the Nuclear Research Center of Grenoble in December 1978 on the CESAR loop. Transient sodium boiling development, dry-out apparition, clad melting, and relocation could be followed.

- 2249 (LA--7938-C, pp 125-140) HYDRODYNAMIC FRAGMENTATION OF DROPS. Patel, P.D.; Theofanous, T.G. (Purdue Univ., West Lafayette, IN). Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

Studies of the morphology of fragmentation of liquid drops in a liquid medium due to shock-induced flows are reported. For high density ratio and high interfacial tension systems, the fragmentation is found to be probably due to the penetration of the drop by unstable waves. An envelope of conditions which characterize drop breakup in these systems has been deduced.

Analytical models, which involve the time constants of both the drop-piercing Taylor instability and the wake-forming shear effects, reflect our experimental results well. In addition, it is found that the mercury-water system studied here should simulate the uranium dioxide/sodium pair, which is of interest in safety studies of fast reactors.

- 2250 (LA--7938-C, pp 141-152) FUEL AND COOLANT MOTIONS FOLLOWING PIN FAILURE: EPIC MODELS AND THE PBE-5S EXPERIMENT. Garner, P.L.; Abramson, P.B. (Argonne National Lab., IL). Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

The EPIC computer code has been used to analyze the post-fuel-pin-failure behavior in the PBE-5S experiment performed at Sandia Laboratories. The effects of modeling uncertainties on the calculation are examined. The calculations indicate that the majority of the piston motion observed in the test is due to the initial pressurization of the coolant channel by fuel vapor at cladding failure. A more definitive analysis requires improvements in calculational capabilities and experiment diagnostics.

- 2251 (LA--7938-C, pp 153-162) ANALYSIS OF TREAT TRANSIENT OVERPOWER EXPERIMENTS USING THE PLUTO CODES. Wider, H.U.; Semenza, L.A. (Argonne National Lab., IL). Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

Pre- and post-test computer analyses of TREAT transient overpower test H6 are presented and compared with an earlier analysis of E8. The computer models used were the PLUTO code and its successor, the PLUTO2 code. A good agreement with several events could be achieved in the post test simulations, which supports the basic modeling in the codes used. The pre-test analyses suffered from the uncertainty in the pin failure sequence. Fission-gas pressures in the pins at failure time were on the order of 15 to 30 MPa in both tests. Plenum fission-gas release may have played an important role in both tests. Although a stronger FCI tended to be promoted by the larger hydraulic forces in the H6 test with its higher pump pressure and flow velocity, the voiding was less pronounced for the same reasons. Fuel sweepout from short EBR-II irradiated pins failing into non-voided channels tended to be on the order of 5 to 20 g of fuel per pin.

- 2252 (LA--7938-C, pp 163-177) FUNDAMENTAL ASPECTS OF MOLTEN CLAD RELOCATION. Marrotte, G.N. (General Electric Corp., San Jose, CA); Theofanous, T.G. Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

Experimental data on the transient motion of a suddenly sheared vertical liquid-metal film and its mechanical interaction with the driving gas flow are presented. New correlations describing these phenomena, and application to high interfacial tension systems are derived from these data and applied to the prediction of clad relocation dynamics in LMFBR core melt accidents.

- 2253 (LA--7938-C, pp 178-184) COMPARISON OF THE PREDICTIONS FROM THE MULTICHANNEL CLAD RELOCATION MODEL TO AVAILABLE IN-PILE EXPERIMENTAL DATA. Ishii, M.; Chen, W.L.;

Kuzay, T.M. (Argonne National Lab., IL). Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

In a hypothetical unprotected undercooling accident in liquid-metal fast breeder reactors, coolant sodium is rapidly heated to boiling. The sodium boiling leads to the dryout and eventual melting of the cladding. The cladding motion upon melting is considered to be important in determining the reactivity effect, blockage formation, and its effect on subsequent fuel motion. A computer code MULCLAD which is based on multi-channel cladding motion model is used for the comparative studies with available experimental data. R4 and R5 TREAT in-pile test and 37 pin P3A test by SLSF Project in the Engineering Test Reactor have been analyzed in detail.

- 2254 (LA--7938-C, pp 185-196) SIMULATION OF CLADDING RELOCATION IN THE TREAT-EXPERIMENT R5 WITH THE CMOT CODE. Angerer, G.; Woll, D. Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

The CMOT code is an analytical tool to simulate cladding motion after melting and its resolidification and blockage formation in voided coolant channels during hypothetical core disruptive accidents in LMFBR's. It was applied to the TREAT experiment R5 to check whether the code is able to correctly reproduce the cladding relocation process. The agreement achieved between experimental and analytical data is encouraging. The results show that cladding relocation may be simulated by the assumption of motion of the molten cladding as a film along the fuel pin surface. The results also show that wave formation and wave motion are important cladding transport mechanisms.

- 2255 (LA--7938-C, pp 197-217) ANALYSIS OF FUEL BEHAVIOUR IN SCARABEE 1ST PHASE EXPERIMENTS. Schmitz, F.; Matthews, J.R. Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

After a short description of the SCARABEE first phase program and of the approach adopted in the interpretation, the main experimental results used for the fuel behaviour analysis are given. Calculation results from the different computer codes are then compared to the experimental results.

- 2256 (LA--7938-C, pp 218-241) SIMMER-II ANALYSIS OF THE R-7 TREAT TEST. Bohl, W.R. Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

The SIMMER-II code was used in a simulation of the R-7 loss-of-flow TREAT test. Comparison with the test scenario encompassed thermocouple data, flowmeter traces, hodoscope data, and the posttest configuration. Reasonably successful results were obtained on the overall heat balance. The calculated boiling time was within 0.2 s of the experimental result, and respectable agreement with the integrated lower flowmeter trace was achieved over the first 2 s of boiling. Although the details of cladding relocation and blockage formation could not be well simulated, the calculated fuel motion appears to be plausible given the available experimental information. It was further concluded that this type of study is quite

useful for revealing areas for model improvement.

- 2257 (LA--7938-C, pp 242-256) ANALYSIS OF TREAT TESTS L7 AND L8 WITH SAS3D, LEVITATE AND PLUTO2. Bowers, C.H.; Tentner, A.M.; Wider, H.U. (Argonne National Lab., IL). Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

Fuel motion in TREAT Tests L7 and L8 is analyzed with the SAS3D, LEVITATE, and PLUTO2 computer codes. These results are compared with fuel motion data obtained from the fuel neutron hodoscope. It is concluded that fission gas acts to disperse fuel during the early stages of fuel motion.

- 2258 (LA--7938-C, pp 257-268) COMPARISON OF FIGAS SWELLING AND GAS RELEASE PREDICTIONS WITH EXPERIMENT. Ostensen, R.W. (Sandia Labs., Albuquerque, NM). Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

FIGAS calculations were compared to fuel swelling data from the FDI tests and to gas release data from the FGR39 test. Late swelling and gas release predictions are satisfactory if vacancy depletion effects are added to the code. However, early swelling predictions are not satisfactory, and early gas release predictions are very poor. Explanation of these discrepancies is speculative.

- 2259 (LA--7938-C, pp 269-279) ANALYSIS OF FUEL DISPERSAL IN DIRECT ELECTRICAL HEATING EXPERIMENTS. Gruber, E.E.; DiMelfi, R.J.; Bandyopadhyay, G. (Argonne National Lab., IL). Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

The effects of transient history on fuel motion driven by fission gas have been investigated in a series of direct-electrical-heating tests on quartz-clad irradiated fuel. Variations in the slow, Stage-I portion of the transient thermal history resulted in pronounced differences in fuel behavior during the subsequent, more rapid, Stage-II portion. Analysis of transient fission-gas behavior, considering both the timing of gas release to grain boundaries and the non-equilibrium character of grain-boundary gas bubbles, provides a coherent picture of the role of fission gas in the observed fuel failure. The results indicate the conditions under which fission gas may efficiently drive expulsion of molten fuel through cracks, as well as conditions under which gross radial swelling of the fuel may be expected. These predictions are compatible with the observed nature of fuel failure.

- 2260 (LA--7938-C, pp 280-300) ANALYSIS OF A PROTOTYPIC AND A SIMULANT EXPERIMENT RELATED TO FREEZING IN FLOW CHANNELS. Hakim, S.J. (Argonne National Lab., IL). Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

An analysis is presented of two experiments, one involving simulant materials, and a second more prototypic experiment, related to freezing of molten LMFBR core material in flow channels. Good agreement is obtained between predictions and experimental results for both experiments. It is argued that simulant experiments could provide a reasonable vehicle for validation of

isolated phenomena related to the freezing process while prototypic experiments provide a more realistic view of the integrated behavior of molten core material during the accident.

- 2261 (LA--7938-C, pp 301-307) MOLTEN FUEL RELOCATION AND CONNECTED PROBLEMS: THE GELCO CODE, COMPARISON WITH EXPERIMENTAL RESULTS, APPLICATION TO REACTOR. de Lapparent, D. Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

The problem of fuel freezing is illustrated in the case of a Hypothetical Core Disruptive Accident. The GELCO Code and an experiment using molten UO_2 are presented. A tentative application is made to Reactor conditions.

- 2262 (LA--7938-C, pp 308-319) SIMMER ANALYSIS OF SRI HIGH PRESSURE BUBBLE EXPANSION EXPERIMENTS. Rexroth, P.E.; Suo-Anttila, A.J. Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

SIMMER-II was used to analyze the results of the SRI nitrogen bubble expansion experiments. Good agreement was found for all of the experiments analyzed as well as the theoretical isentropic limiting case. Scaling to a full size CRBR reactor reveals no significant scaling effects for the structureless core.

- 2263 (LA--7938-C, pp 320-334) TERMINATION PHASE OF CORE DISRUPTIVE ACCIDENTS IN LMFBRs. Saito, M.; Theofanous, T.G. Jul 1979.

Specialists' workshop on predictive analysis of material dynamics in LMFBR safety experiments, March 13-15, 1979.

The most significant finding of this work is that the presence of entrainment during bubble growth was found significantly less than expected on the basis of the current literature. A small amount of entrainment is seen as the impact is approached and a new entrainment mechanism is suggested. Additional study of this region will be required to predict the extent of entrainment and quantify its impact on energy release.

- 2264 (LA--7975-PR) EXAMINATION OF FAST REACTOR FUELS AND FBR ANALYTICAL QUALITY ASSURANCE STANDARDS AND METHODS. PROGRESS REPORT, APRIL 1-JUNE 30, 1979. Maraman, W.J. (comp.). (Los Alamos Scientific Lab., NM (USA)). Aug 1979. Contract W-7405-ENG-36. 16p. Dep. NTIS, PC A02/MF A01.

This project is directed toward the examination and comparison of the effects of neutron irradiation on Liquid Metal Fast Breeder Reactor (LMFBR) Program fuel materials. Characterizations of unirradiated and irradiated fuels by analytical chemistry methods will continue, and additional methods will be modified and mechanized for hot-cell application. Macro- and micro-examinations will be made on fuel and cladding, using the shielded electron microprobe, emission spectrograph, radiochemistry gamma scanner, mass spectrometers, and other analytical facilities. New capabilities will be developed in gamma scanning analyses to assess spatial distributions of fuel and fission products, mass spectrometric measurements of burnup and fission gas constituents, and other chemical analyses. Reliable chemical characterizations are essential to the development of high-quality fuels, control rods, and other reactor components in the FBR Program. Tasks were

designed to assure the quality of a chemical characterizations necessary to evaluate reactor components relative to specifications. Tasks include: (1) the continual preparation and distribution of carefully characterized calibration materials and quality control samples for use in quality assurance surveillance programs of vendor and purchaser analytical chemistry laboratory operations, (2) the development of and the guidance in the use of quality assurance programs for sampling and analysis, (3) the development of improved methods of analysis, (4) the preparation of continuously updated analytical method manuals, and (5) maintenance of referee analysis capabilities to characterize, on a reimbursable basis, materials in dispute between fabricators (vendors) and reactor operators (purchasers). At this time, the emphasis is on the FFTF. The program will be extended, as appropriate, to the LMFBR demonstration and large production facilities in the future.

- 2265 (LA--8132=PR) EXAMINATION OF FAST REACTOR FUELS AND FBR ANALYTICAL QUALITY ASSURANCE STANDARDS AND METHODS. PROGRESS REPORT, JULY 1-SEPTEMBER 30, 1979. Maraman, W.J. (comp.). (Los Alamos Scientific Lab., NM (USA)). Nov 1979. Contract W-7405-ENG-36. 17p. Dep. NTIS, PC A02/MF A01.

This is the twentieth quarterly report on the Examination of Fast Reactor Fuels and Fast Breeder Reactor (FBR) Analytical Quality Assurance and Methods at the Los Alamos Scientific Laboratory. Most of the investigations discussed are of the continuing type. Results and conclusions described may therefore be changed or augmented as the work continues. Published reference to results cited in this report should not be made without obtaining explicit permission to do so from the person in charge of the work.

- 2266 (LA--8238=PR) EXAMINATION OF FAST REACTOR FUELS AND FBR ANALYTICAL QUALITY ASSURANCE STANDARDS AND METHODS. PROGRESS REPORT, OCTOBER 1-DECEMBER 31, 1979. Maraman, W.J. (comp.). (Los Alamos Scientific Lab., NM (USA)). Feb 1980. Contract W-7405-ENG-36. 12p. Dep. NTIS, PC A02/MF A01.

This project is directed toward the examination and comparison of the effects of neutron irradiation of Liquid Metal Fast Breeder Reactor (LMFBR) Program fuel materials. Unirradiated and irradiated materials will be examined as requested by the Reference Fuels System Branch of the Division of Reactor Research and Technology (DRRT). Reliable chemical characterizations are essential to the development of high-quality fuels, control rods, and other reactor components in the FBR Program. The objective tasks are designed to assure the quality of the chemical characterizations necessary to evaluate reactor components relative to specifications.

- 2267 (LA--8292=MS) EFFECT OF MODEL CHOICE AND SAMPLE SIZE ON STATISTICAL TOLERANCE LIMITS. Duran, B.S.; Campbell, K. (Los Alamos Scientific Lab., NM (USA)). Mar 1980. Contract W-7405-ENG-36. 50p. Dep. NTIS, PC A03/MF A01.

Statistical tolerance limits are estimates of large (or small) quantiles of a distribution, quantiles which are very sensitive to the shape of the tail of the distribution. The exact nature of this tail behavior cannot be ascertained from small samples, so statistical tolerance limits are frequently computed using a statistical model chosen on the basis of theoretical

considerations or prior experience with similar populations. This report illustrates the effects of such choices on the computations.

- 2268 (LA-UR--77-2378) FINITE-DIFFERENCE METHODS IN MULTI-DIMENSIONAL TWO-PHASE FLOW. Travis, J.R. (Los Alamos Scientific Lab., NM (USA)). 1977. Contract W-7405-ENG-36. 19p. (CONF-771204--1). Dep. NTIS, PC A02/MF A01.

From 3. international symposium on computing methods in applied sciences and engineering; Versailles, France (5 Dec 1977).

In the summer of 1974, the Theoretical Division of the Los Alamos Scientific Laboratory began several research programs in the area of reactor safety for the United States Nuclear Regulatory Commission. Research efforts were started in the Liquid Metal Fast Breeder (LMFBR) and the Light Water Reactor (LWR) safety programs. The character of the Theoretical Division was to develop computer codes for the safety analysis of these reactor systems. The question of whether or not, during the course of a hypothetical accident sequence in an LMFBR, the core will subside to a coolable configuration without secondary critical bursts has never been resolved. To aid the study of this question, a computer program called SIMMER (S/sub N/, Implicit, Multified, Multicomponent, Eulerian Recriticality) was to be developed to predict the dynamics of extreme hypothetical accident sequences during which extended core motion is expected. This time-dependent computer code called for combining an advanced multidimensional, multiphase fluid dynamic methodology with multidimensional neutron transport theory and improved equation-of-state technology. In the LWR program, the research emphasis was to push forward in two areas: (1) the development of advanced multiphase fluid dynamic methods and computer programs for performing basic research and analyzing areas in thermal hydraulics important to the safety of water reactors, and (2) the development of an advanced 'best estimate' systems code called TRAC (Transient Reactor Analysis Code) for analyzing loss-of-coolant accidents and anticipated-transients-without-scrum in light water reactors.

- 2269 (LA-UR--78-495) ROLE OF MECHANISTIC ANALYSIS IN PROVIDING PAHR INITIAL CONDITIONS. Jackson, J.F. (Los Alamos Scientific Lab., NM (USA)). 1977. Contract W-7405-ENG-36. 3p. (CONF-771108--9). Dep. NTIS, PC A02/MF A01.

From Post accident heat removal information exchange meeting; Chicago, IL, USA (2 Nov 1977).

The objective of this paper is to discuss how Postaccident Heat Removal (PAHR) calculations interface with the earlier accident phases in overall mechanistic accident analyses, to briefly review the evolution of the transition phase as an important path in LMFBR core disruptive accidents, and to assess the impact of the transition phase on PAHR analyses.

- 2270 (LA-UR--78-1265) TRANSPORT PROCESSES IN LMFBR CORE DISRUPTIVE ACCIDENTS. Boudreau, J.E.; Smith, L.L. (Los Alamos Scientific Lab., NM (USA)). 1978. Contract W-7405-ENG-36. 16p. (CONF-780918--2). Dep. NTIS, PC A02/MF A01.

From International centre for heat and mass transfer; Dubrovnik, Yugoslavia (4 Sep 1978).

In this paper, a few of the important problem areas in LMFBR safety are defined and the results of related SIMMER calculational results are outlined. The key transport

phenomena that are known have been identified and the need for specific experimental/analytical research has been discussed. The SIMMER code is in its early stages of developmental verification, and the need for a comprehensive experiment analysis activity is clear. Experimental research should include a balanced set of tests that lend insight both in the isolated phenomena as well as span the range of interest in interactive, high temperature transient behavior. If uncertainty bands for the transport phenomena can be quantified, sensitivity studies can be performed to evaluate the adequacy of our current state of knowledge, and conversely the need for further research.

- 2271 (LA-UR--78-2602) ANALYSIS OF TERMINATED TOP ACCIDENTS IN THE FTR USING THE LOS ALAMOS FAILURE MODEL. Mast, P.K.; Scott, J.H. (Los Alamos Scientific Lab., NM (USA)). 1978. Contract W-7405-ENG-36. 11p. (CONF-781022--29). Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

A new fuel pin failure model (the Los Alamos Failure Model), based on a linear life fraction rule failure criterion, has been developed and is reported herein. Excellent agreement between calculated and observed failure time and location has been obtained for a number of TOP TREAT tests. Because of the nature of the failure criterion used, the code has also been used to investigate the extent of cladding damage incurred in terminated as well as unterminated TOP transients in the FTR.

- 2272 (LA-UR--78-2650) REMOTE EXAMINATION OF SHROUD TUBES IN LMFBR FUEL ELEMENTS. Ledbetter, J.M.; Serna, O.; Nicol, A.G.; Wood, W.T. (Los Alamos Scientific Lab., NM (USA)). 12 Nov 1978. Contract W-7405-ENG-36. 12p. (CONF-781105--52). Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

A system for providing a means of remote ejection and examination of sodium bonded shroud encapsulated fuels has been in use at LASL for over a year. The system is remotely capable of precise machining for clad separation and splitting at specified areas. Controlled heating and constant temperature monitoring is incorporated for sodium melting. The system provides for numerous operations such as: visual examinations, measurements, photography, and sodium dissolution.

- 2273 (LA-UR--78-2711) EVALUATION OF LMFBR FUEL-MOTION DIAGNOSTICS INSTRUMENTATION WITH PARKA. Evans, A.E. Jr.; Orndoff, J.D.; Talbert, W.L. Jr. (Los Alamos Scientific Lab., NM (USA)). 1978. Contract W-7405-ENG-36. 10p. (CONF-781033--15). Dep. NTIS, PC A02/MF A01.

From IEEE nuclear science symposium; Washington, DC, USA (18 Oct 1978).

To aid in the design of LMFBR safety test experiments and safety test facilities (STF), a program of evaluation of concepts for fuel-motion diagnostics instrumentation has been undertaken. A part of this evaluation is being done at PARKA, a Rover project critical assembly which has been modified to study the self nuclear image from driven FTR-type fuel assemblies. Feasibility of obtaining fast-neutron images of single-pin voids in assemblies of up to 127 fuel pins has been demonstrated, albeit marginally for the larger fuel bundles. The feasibility of using in-core detectors as fuel-motion monitors has also been studied. Use of PARKA in a pulsed mode to study

STF transient phenomena is discussed.

- 2274 (LA-UR--78-2932) IRRADIATION PERFORMANCE OF HELIUM-BONDED URANIUM--PLUTONIUM CARBIDE FUEL ELEMENTS. Latimer, T.W.; Petty, R.L.; Kerrisk, J.F.; DeMuth, N.S.; Levine, P.J.; Boltax, A. (Los Alamos Scientific Lab., NM (USA)). 1979. Contract W-7405-ENG-36. 12p. (CONF-790306--3). Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

The current irradiation program of helium-bonded uranium--plutonium carbide elements is achieving its original goals. By August 1978, 15 of the original 171 helium-bonded elements had reached their goal burnups including one that had reached the highest burnup of any uranium--plutonium carbide element in the U.S.--12.4 at.%. A total of 66 elements had attained burnups over 8 at.%. Only one cladding breach had been identified at that time. In addition, the systematic and coordinated approach to the current steady-state irradiation tests is yielding much needed information on the behavior of helium-bonded carbide fuel elements that was not available from the screening tests (1965 to 1974). The use of hyperstoichiometric (U,Pu)C containing approx. 10 vol% (U,Pu)₂C₃ appears to combine lower swelling with only a slightly greater tendency to carburize the cladding than single-phase (U,Pu)C. The selected designs are providing data on the relationship between the experimental parameters of fuel density, fuel-cladding gap size, and cladding type and various fuel-cladding mechanical interaction mechanisms.

- 2275 (LA-UR--78-2933) DESIGN AND PERFORMANCE OF SODIUM-BONDED URANIUM--PLUTONIUM CARBIDE FUEL ELEMENTS. Kerrisk, J.F.; DeMuth, N.S.; Petty, R.L.; Latimer, T.W.; Vitti, J.A.; Jones, L.J. (Los Alamos Scientific Lab., NM (USA)). 1979. Contract W-7405-ENG-36. 13p. (CONF-790306--4). Dep. NTIS, PC A02/MF A01.

From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979).

Recent results from irradiation tests indicate that sodium-bonded elements provide a practical advanced fuel element design for use in LMFBRs. Shroud tubes have effectively controlled fuel-cladding mechanical interaction; thicker and stronger claddings have also been effective in this respect. Burnups to 11 at.% have been achieved under typical operating conditions. A heterogeneous core with a breeding ratio of 1.55 and a compound system doubling time of less than 13 years has been designed using these element designs.

- 2276 (LA-UR--79-162) APPROXIMATE INVERSION METHOD FOR FIVE POINT DIFFERENCE MATRIX EQUATIONS. Steinke, R.B. (Los Alamos Scientific Lab., NM (USA)). 1979. Contract W-7405-ENG-36. 12p. (CONF-790402--43). Dep. NTIS, PC A02/MF A01.

From Computational methods in nuclear engineering; Williamsburg, VA, USA (23 Apr 1979).

SIMMER, a best estimate computer program for LMFBR disrupted core analysis, is being developed at the Los Alamos Scientific Laboratory. Recent fluid dynamics methods development for SIMMER was concerned with enhancing numerical stability and reducing calculational effort. One such development considered the evaluation of the pressure change and material density change

distributions during a time step. The spatially uncoupled approximate solution to be iterated was replaced with a correct spatially coupled simultaneous solution. This resulted in decreased calculational effort per time step as well as enhanced stability to allow much larger time steps to be taken. This improvement was realized by formulating the correct five diagonal matrix equations in two-dimensional rectilinear geometry and then solving them with an efficient matrix inversion routine. Initially, the Successive Line OverRelaxation (SLOR) method was used to solve these matrix equations. Instances of weakly diagonally dominant matrices requiring hundreds of SLOR iterations to invert them prompted consideration of a more efficient method. A new Approximate Inversion Method (AIM) was felt to have this efficiency. The method is a variation of the direct Crout--Cholesky forward elimination and backward substitute method. Crout--Cholwesky method element operations on a tridiagonal element matrix are applied in AIM as submatrix block operations on a tridiagonal block matrix. To make the method efficient, a submatrix collapsing approximation is applied. Making this approximation requires that the method be iterative. A description of this iterative algorithm and its approximate matrix inversion is included.

- 2277 (LA-UR--79-770) SIMMER ANALYSIS OF SRI HIGH PRESSURE BUBBLE EXPANSION EXPERIMENTS. Rexroth, P.E.; Suo-Anttila, A.J. (Los Alamos Scientific Lab., NM (USA)). 1979. Contract W-7405-ENG-36. 12p. (CONF-790310--3). Dep. NTIS, PC A02/MF A01.

From Specialists workshop on predictive analysis of material dynamics in LMFBR safety expts; Los Alamos, NM, USA (13 Mar 1979).

SIMMER-II was used to analyze the results of the SRI nitrogen bubble expansion experiments. Good agreement was found for all of the experiments analyzed as well as the theoretical isentropic limiting case. Scaling to a full size CRBR reactor reveals no significant scaling effects for the structureless core.

- 2278 (LA-UR--79-811) MEASUREMENT AND ANALYSIS OF DRAG IN MULTIPHASE FLOW SYSTEMS. Rexroth, P.E.; Starkovich, V.S. (Los Alamos Scientific Lab., NM (USA)). 1979. Contract W-7405-ENG-36. 16p. (CONF-790423--7). Dep. NTIS, PC A02/MF A01.

From 2. multiphase flow and heat transfer symposium workshop; Miami Beach, FL, USA (16 Apr 1979).

A combined experimental and analytical program undertaken to evaluate an interfield momentum coupling model used in the computer code SIMMER is described. The behavior of a slugging, vapor-particle flow system was observed and recorded using gamma densitometry, differential pressure measurement, and motion pictures. The primary parameter observed was slug period. When the system was modeled using SIMMER, the calculated behavior of the flow was qualitatively similar to that observed experimentally, but both the period and maximum slug height were underestimated, indicating too weak coupling between the vapor and particle fields. The SIMMER drag correlation was modified, resulting in much better agreement. Final discrepancies between experiment and analysis are discussed.

- 2279 (LA-UR--79-1230) HEAT TRANSFER AND THERMAL LOSSES IN ABOVE-CORE REGIONS. Bell, C.R.; Boudreau, J.E. (Los Alamos Scientific Lab., NM (USA)). 1979. Contract W-7405-ENG-36. 19p. (CONF-790810--1). Dep. NTIS, PC

A02/MF A01.

From 2. international seminar on containment of fast breeder reactors; Berlin, F.R. Germany (9 Aug 1979).

Heat transfer and thermal losses in above-core regions can substantially influence the conversion of thermal energy to damage potential following postulated core-disruptive accidents in a liquid metal fast breeder reactor. Any assessment of these influences is complex and difficult to substantiate because of the transient, interactive environment in which they occur. A perspective is developed for the role of heat-transfer processes in the context of fluid/structure dynamics, containment strength, accident severity, sodium involvement, and uncertainties in current knowledge. The Clinch River Breeder Reactor (CRBR) is used as an example to quantify this perspective. Containment limits are estimated for CRBR to allow a comparative evaluation of the containment tolerance as different physical effects are considered.

- 2280 (LA-UR--79-1874) OVERVIEW AND STATUS OF THE SIMMER TESTING PROGRAM. Scott, J.H. (Los Alamos Scientific Lab., NM (USA)). 1979. Contract W-7405-ENG-36. 13p. (CONF-790816--16). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Los Alamos Scientific Laboratory has undertaken an extensive experiment analysis program to test the results of SIMMER Liquid Metal Fast Breeder Reactor (LMFBR) accident calculations. Initially, we will test the postdisassembly work-energy partition problem. The SIMMER-calculated order-of-magnitude reduction of available kinetic energy following a severe hypothetical core-disruptive accident (HCDA) can be attributed to (1) purely fluid-dynamic effects; and (2) rate-controlled effects, such as phase transitions and heat transfer. We have chosen to test separately each class of mitigator. In this paper we review the experiments initially chosen for testing of each class of mitigator and report on the status of the analyses. We enumerate several problems in SIMMER that experiment analysis has disclosed. Finally, needs for future experiments are discussed.

- 2281 (LA-UR--79-1918) SIMMER-II CODE AND ITS APPLICATIONS. Smith, L.L. (Los Alamos Scientific Lab., NM (USA)). 1979. Contract W-7405-ENG-36. 11p. (CONF-790816--13). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The significant features of SIMMER-II, a disrupted-core analysis code, are described. The code has the capabilities to begin space-time neutronics calculations from nonstationary reactor states, to track the intermixing of fuel of different enrichments, and to model the complicated heat- and mass-transfer processes that occur in the transition phase. Example calculations are presented for analyses of whole-core accidents and for analyses of experiments supporting the code models.

- 2282 (LA-UR--79-1956) ROLE OF SIMILITUDE IN THE DESIGN OF LMFBR SAFETY-RELATED SIMULATION EXPERIMENTS. Chapyak, E.J.; Starkovich, V.S. (Los Alamos Scientific Lab., NM (USA)). 1979. Contract W-7405-ENG-36. 11p. (CONF-790816--15). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug

1979).

In this paper we examine the implications of scaling in the design of simulation experiments for two major areas of fast reactor safety research - the transition phase and postdisassembly energetics. Specific scaling requirements and compatible experimental designs are formulated and compared with ongoing programs having similar objectives. Suggestions are also outlined for future out-of-pile experimental research.

- 2283 (LA-UR--79-2037) DEVELOPMENTS IN THE LASL FUEL PIN IMAGING SYSTEM: PINEX-3A. Lumpkin, M.H.; Berzins, G.J.; Cosimi, R.A.; D'Hare, T.E.; Davidson, J.R. (Los Alamos Scientific Lab., NM (USA)). 1979. Contract W-7405-ENG-36. 11p. (CONF-790816--14). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The LASL Fuel Pin Imaging System was evaluated using a series of 10 TREAT transients, each of approx. 240-MW peak power. HEDL provided the fuel-ejection type capsule with annular fuel pellets. The pin visibility threshold was determined to be approx. 20-MW of TREAT power (approx. 130 W/g), almost an order of magnitude improvement over our PINEX-2 threshold. The impact of changes in instrumentation, imaging apertures, and fluors that produced the improved sensitivity are reported. Results of a time-integrated imaging technique are also presented.

- 2284 (LA-UR--79-2141) INFLUENCE OF TREAT NONPROTOTYPIC EFFECTS ON PREDICTION OF PIN FAILURE IN LMFBRs. Mast, P.K.; Scott, J.H. (Los Alamos Scientific Lab., NM (USA)). 1979. Contract W-7405-ENG-36. 12p. (CONF-790816--41). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The effect of the nonprototypic stagnant coolant environment in the static-capsule TOP TREAT tests has been investigated using the LAFM code. The cladding temperature was shown to be higher relative to the fuel temperature for the simulated 50 cents/s transients than in a hypothetical 50 cents/s FTR transient. This higher relative cladding temperature significantly decreases the importance of fuel-cladding differential thermal expansion as a failure mechanism and causes concern about the direct applicability of the 50 cents/s static-capsule TREAT test results to hypothetical FTR transients.

- 2285 (LA-UR--79-2245) INTEGRAL POSTDISASSEMBLY ENERGETICS TEST: A PRELIMINARY INVESTIGATION. Barts, E.W.; Asprey, M.W.; Burns, R.D. III; Chapyak, E.J.; Dunwoody, W.E.; Helmick, H.H. (Los Alamos Scientific Lab., NM (USA)). 1979. Contract W-7405-ENG-36. 12p. (CONF-790816--44). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Scaled, real-material integral tests could demonstrate mitigating mechanisms in postdisassembly energetic behavior during a hypothetical accident in a liquid metal fast breeder reactor (LMFBR). Such a test series could provide key data for accident analysis codes and LMFBR design and licensing. Studies were performed to determine important system parameters including the test geometry necessary to maintain prototypicality.

Neutronic behavior of the core and radial blanket configuration was studied to assure a controlled energetic burst. An engineering design study was performed to determine test feasibility.

- 2286 (LA-UR--79-2285) EXAMINATION OF SUBASSEMBLY SCALE FUEL MOTION USING THE SIMMER-II CODE. Alcouffe, R.E.; Henninger, R.J. (Los Alamos Scientific Lab., NM (USA)). 1979. Contract W-7405-ENG-36. 13p. (CONF-790865--1). Dep. NTIS, PC A02/MF A01.

From ANS/ENS international meeting on fast reactor safety technology; Vancouver, Canada (19 Aug 1979).

Subassembly scale motion during the early fuel dispersal and extended fuel motion phases of an LMFBR loss of flow accident are examined with the SIMMER-II code. Results are compared and contrasted with previous analyses and selected experiments. The early fuel motion with a SIMMER-II model is calculated with the results given by the SLUMPY module of the SAS3A code. SLUMPY may greatly overpredict the extent and magnitude of fuel motion on the time scale of interest using the nominal SLUMPY parameters. By comparing with pin geometry fuel freezing experiments, SIMMER-II is calibrated to calculate the experimental fuel penetration distances.

- 2287 (LA-UR--79-2340) SOME RECRITICALITY STUDIES WITH SIMMER-II. Bohl, W.R. (Los Alamos Scientific Lab., NM (USA)). 1979. Contract W-7405-ENG-36. 12p. (CONF-790816--38). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

The SIMMER-II code was applied to the problem of evaluating the potential for recriticality in an LMFBR HCDA. The phenomenology examined was based on the post SAS3D behavior in a CRBR LOF accident. The SIMMER-II results were found to be sensitive to the development of fuel blockages. If blockages were formed close to the core, the core-disruption phase of the behavior resulted from the significant energy required following SAS3D termination to raise the average steel temperature to saturation conditions; also material deposition near heat sinks eliminated condensation surfaces and induced core pressurization and fuel collapse. Excessive ramp rates (greater than 100/s) were judged to be unlikely, but improved knowledge and calculational treatments of the applicable phenomenology would be desirable.

- 2288 (LA-UR--80-1159) DEEP PENETRATION CALCULATIONS. Thompson, W.L.; Deutsch, D.L.; Booth, T.E. (Los Alamos Scientific Lab., NM (USA)). Apr 1980. Contract W-7405-ENG-36. 29p. (CONF-800471--1). NTIS, PC A03/MF A01.

From Seminar on theory and applications of Monte Carlo methods; Oak Ridge, TN, USA (Apr 1980).

Several Monte Carlo techniques are compared in the transport of neutrons of different source energies through two different deep-penetration problems each with two parts. The first problem involves transmission through a 200-cm concrete slab. The second problem is a 90° bent pipe jacketed by concrete. In one case the pipe is void, and in the other it is filled with liquid sodium. Calculations are made with two different Los Alamos Monte Carlo codes: the continuous-energy code MCNP and the multigroup code MCMG.

2289 (LA-UR--80-1283) THERMAL CONDUCTIVITIES OF MIXED CARBIDE FUEL AND BLANKET MATERIALS. Lewis, H.D. (Los Alamos Scientific Lab., NM (USA)). May 1980. Contract W-7405-ENG-36. 13p. (CONF-800532--1). NTIS, PC A02/MF AJ1.

From Core components working group information meeting; Richland, WA, USA (13 May 1980).

Measurement of the thermal and electrical transport properties of the uranium-plutonium carbides began in 1950, the bulk of the experimental work being performed in the US, UK, and France. During 1974 and 1975, as part of the national LMFBR program effort to provide unified design parameters, a series of critical reviews on the thermodynamic, physical, chemical, and mechanical properties of uranium-plutonium carbide (and nitride) fuels was prepared by authors at several US laboratories. Property values recommended by these reviews were compiled in ANL-APP-27. The LASL review on the transport properties of uranium and plutonium carbides recommended the conductivity values that are presented.

2290 (LMC--77-2) CLS/DHX MODULE TESTING IN SPTF. FINAL REPORT. Archbold, P. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 30 Sep 1977. Contract EY-76-C-03-0700. 81p. AT.

This report covers testing of the Dump Heat Exchanger (DHX) installed in the Sodium Pump Test Facility (SPTF) at the Liquid Metal Engineering Center, Santa Susana, California. The overall objectives of the test program were to obtain data on heat transfer capability and instrumentation and controls associated with the heat exchanger for comparison against the models for the Closed Loop System (CLS) Dump Heat Exchanger (DHX) to be installed for the Fast Flux Test Facility (FFTF) at Hanford, Washington. Tests were performed on a noninterfering basis during the FFTF Prototype Pump Test Program. Early data on the heat transfer capability and preheat capability of the DHX indicated that the performance was somewhat lower than expected. Added insulation and rework of the baffle in the header box resulted in improved performance. Instrumentation and controls did not perform as expected and results indicate a need for reevaluation of the CLS/DHX control systems. Comparative tests for the flowmeter indicated the readings to be approximately 20 percent low.

2291 (LMC--77-3) CALIBRATION OF THE FFTF LEVEL MEASURING SYSTEMS. Fischer, B.E. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 1 Dec 1977. Contract EY-76-C-03-0700. 447p. AT.

This report summarizes the results of a program conducted at the Liquid Metal Engineering Center (LMC) to calibrate all of the FFTF inductive level measuring systems manufactured by Westinghouse Industrial and Government Tube Division. Sixty-two level measuring systems were calibrated during the program. Performance data was collected at sodium temperatures between 300 and 1200°F (149 and 649°C). Five of the sixty-two systems were calibrated in a NaK test facility at temperatures between 70 and 500°F (21.1 and 260°C).

2292 (LMC--77-4) FFTF PROTOTYPE PUMP TESTING IN SPTF. FINAL REPORT. Frangos, A.Z. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 15 Oct 1977. Contract EY-76-C-03-0700. 463p. AT.

In-sodium testing is described for the full-size Fast Flux Test Facility (FFTF) prototype pump in the Sodium Pump Test Facility (SPTF) at the Liquid Metal Engineering Center (LMC) over the complete range of steady-state hydraulic and thermal conditions as defined by Westinghouse-Hanford, the test requester. The overall objectives of the prototype pump test program were to demonstrate the highest possible confidence in the design of the pump prior to its actual operation in the FFTF plant. Test results reported include baseline and steady-state operation, thermal transient testing, hydraulic performance in sodium, endurance testing at steady-state conditions, cavitation tests, and assembly and disassembly procedures. A chronology for the test program is provided including a complete listing of test procedures, specifically the Special Test Procedures under which a major portion of the initial testing was performed.

2293 (LMC--77-5) TESTING OF THE FFTF PLANT PROTECTION SYSTEM PRESSURE TRANSDUCERS. Twa, G.J. (Dames and Moore, Los Angeles, CA (USA)). 15 Jan 1978. Contract EY-76-C-03-0700. 120p. AT.

The testing of three NaK transmission type, high-temperature pressure transducers is described. These transducers are identical to those procured for the FFTF Plant Protection System. Tests were conducted out of sodium to evaluate the performance characteristics and to determine if they meet the requirements of the procurement specification. The transducers were evaluated at process seal temperatures to 1200°F (649°C). Time response to step changes in pressure was performed to verify that these units met the 50-ms requirements. In-sodium tests were conducted to determine the effect of long-term exposure to sodium and to evaluate the long-term stability of the transducers. The transducers were tested for approximately 16,000 h in sodium with more than 10,000 h of this time at 1200°F (649°C).

2294 (LMC--77-6) MESHLESS COLD TRAP A PERFORMANCE EVALUATION BY TEST AND ANALYSIS. Goodman, L. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 5 Apr 1979. Contract EY-76-C-03-0700. 53p. AT.

Cold trap development has been hindered by an inadequate understanding of the basic phenomena involved. Consequently, there has been minimal guidance for designing a cold trap to provide a specified performance. Operational requirements at the Liquid Metal Engineering Center dictated a need for specified performance design. A full-scale, cold trap test was conducted to validate an analytical model for trap performance and to prove the effectiveness of traps without internal mesh. The test was successful on both counts.

2295 (LMC--77-7) SAMPLING AND CHARACTERIZATION OF SODIUM-WATER REACTION PRODUCTS. Eichelberger, R.L. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 15 Dec 1977. Contract EY-76-C-03-0700. 46p. Dep. NTIS, PC A03/MF A01.

The products of sodium-water reactions which had taken place in the Large Leak Test Rig at the Liquid Metal Engineering Center were sampled and characterized. The products consisted of the expected compounds in the Na-O-H system, and dissolved readily in ethanol. Removal of such products from a system by melting and draining does not appear feasible, but removal by dissolution in alcoholic solvents should be investigated further.

2296 (LMEC--77-8) DEMONSTRATION OF CONTACT AND PLUGGING TUBE TYPE LEAK DETECTOR PERFORMANCE IN FFTF REACTOR INLET GUARD PIPE. Klea, J.A. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 31 Jan 1978. Contract EY-76-C-03-0700. 87p. AT.

The performance (response) of a contact and plugging tube type leak detector was evaluated in a heated and inerted pipe mockup of the reactor inlet piping system in the vicinity of the inlet nozzle and guard vessel annulus. Other than an initial test at 600°F (315.6°C) and the last test at 400°F (204.4°C), the tests were run at 680°F (360°C) with 1 percent O_2 and 1000 ppm (vol) moisture. A leak rate system was designed to permit liquid sodium to leak from the inside radius surface of the inlet pipe onto the bottom of the guard pipe. Leak rates were varied in the 50- to 800-g/h range. The behavior and nature of the sodium was evaluated and photographed by disassembling the pipe mockup following each test.

2297 (LMEC--78-1) LIQUID METAL ENGINEERING CENTER (LMEC). ANNUAL TECHNICAL PROGRESS REPORT, JANUARY--DECEMBER 1977. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 15 Apr 1978. Contract EY-76-C-03-0700. 350p. AT.

The immediate objective of the LMEC is to support the LMFBR program by providing sodium component and instrumentation test facilities. In the past, emphasis has been in support of the FFTF. At this time, more and more effort is being channeled into base technology programs in support of LMFBR requirements beyond FFTF. This report is divided into two parts, one dealing with testing initiated by external organizations, the other with LMEC-initiated programs required to improve the test facilities or to meet special test requirements. Only those projects considered to be of technical interest to the LMFBR community are reported.

2298 (LMEC-TDR--78-3) EXPERIMENTAL RESULTS OF SODIUM-WATER REACTION TEST NO. 6. Miller, A.E.; Neely, H.H. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 6 Feb 1978. Contract EY-76-C-03-0700. 511p. AT.

The sixth (SWR-6) and final sodium-water reaction test of Series I was performed in the Large Leak Test Rig (LLTR) on November 3, 1977. Presented are computer-produced plots of the transient data obtained and the results of the post-test, nondestructive examinations and cleaning of the test article, Atomic International Modular Steam Generator (AI-MSG) and test system. In the five previous tests (SWR-1 through SWR-5), a double-ended guillotine (DEG) rupture was induced in a standard water tube of the AI-MSG. For SWR-6, a special rupture tube with three times the flow area (3-DEG) was installed in the hockey stick of the AI-MSG. Except for the larger H_2O injection rate (approximately 10.5 lbs/sec vs. approximately 4 lbs/sec) resulting from the larger flow areas (of the rupture tube and connecting injection circuit), SWR-6 and SWR-4 conditions essentially were identical. Superheated steam (approximately 700°F/1900 psig) was injected for 3 seconds into sodium (800°F/100 psig) in the hockey stick region of the AI-MSG, 1.75 inches from the upper tube sheet.

2299 (LMEC-TDR--78-4) AI SUBSCALE INDUCER TEST ARTICLE: FIRST AND SECOND 300 HOUR ENDURANCE TESTS. McDowell, M.W. (Liquid Metal Engineering Center, Canoga Park, CA

(USA)). 14 Feb 1978. Contract EY-76-C-03-0700. 41p. AT.

This report covers the results and associated incidents from the testing of the AI Subscale Inducer Test Article between September 1977 and January 7, 1978. Testing was performed in 750°F sodium at LMEC's Small Component Test Loop (SCTL). The tests conducted during this period were two 300 hour endurance runs. Test conditions were the same for each endurance run. Operation was performed near design speed with Q/N between .80 and .90 and NPSH between 16 and 18 feet. The pump was installed in the facility before each test and removed after each test.

2300 (LMEC-TDR--78-5) STRESSCOAT TEST OF 6-INCH PIPE CLAMP IN TTF. Rollins, C.T. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 15 Feb 1978. Contract EY-76-C-03-0700. 47p. AT.

This report describes the Stresscoat evaluation of the stress distribution in the clamp and adjacent pipe when subjected to clamp bolt load, tensile and compressive load on the clamp normal to the pipe. The Stresscoat evaluation was effective in determining the approximate stress distributions and the maximum stress locations for each type of loading. Based on these results detailed strain gage locations for subsequent testing were selected and presented.

2301 (MLM-MU--77-70-0004) AEROSOL CHARACTERIZATION FROM A SIMULATED HDCA. ANNUAL REPORT. NRC PROGRAM A1171-2. Johnson, E.W.; Fleming, D.L.; Zielinski, R.E.; Zanotelli, W.A. (Mound Lab., Miamisburg, OH (USA)). 27 Oct 1977. Contract EY-76-C-04-0053. 30p. Dep. NTIS, PC A03/MF A01.

Studies were performed to determine the existence of triiodine uranium (V) oxide (Na_3UO_4), since this compound's thermodynamic properties are similar to the compound $Na_3(U,Pu)O_4$ supposedly formed in a HCDA. Although the actual conditions of heat and inert atmosphere were not completely duplicated, a general idea of the compounds formed and their behavior were assessed. The study indicated that the major components formed from a reaction of sodium, uranium, and stainless steel in an air atmosphere were β -sodium metauranate (β - Na_2UO_4) and sodium diuranate ($Na_2U_2O_7$), rather than trisodium uranium (V) oxide (Na_3UO_4). The compound tetrasodium uranyl carbonate $Na_4UO_2(CO_3)_3$ was the major compound formed when the pellets were heated in an argon atmosphere. Although one exception was encountered, experimental evidence indicates that stainless steel does not react with sodium or uranium during a HCDA.

2302 (MP--45-SA-002) REMOVAL OF SODIUM/SODIUM WATER REACTION PRODUCTS FROM THE MSG BY REACTION WITH ORGANIC SOLVENTS. Stearns, J.D. (Energy Technology Engineering Center, Canoga Park, CA (USA)). Dec 1978. Contract EY-76-C-03-0700. 16p. AT.

This document describes the steps required to remove sodium and sodium-water reaction products from the Modular Steam Generator (MSG) following testing in Large Leak Test Rig (LLTR).

2303 (MRR--160, pp 511-529) COMPUTERS PROCESSING THE TEMPERATURE IN THE CORE OF PHENIX. Grisolle, J.; Quenee, R. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)). Jul 1976. (In French). From IAEA/NPPCI specialists meeting on use

of computers for protection systems and automatic control; Neuherberg/Munchen, F.R. Germany (11 May 1976).

The operational reliability of industrial mini-computers justifies their application in reactor protection systems. Taking into consideration the conclusions drawn from experience, improvements have been made on the installation for monitoring core temperatures (TRCT) at the Phenix reactor resulting in the fact that during the last six months of operation no failures have been detected.

- 2304 (MRR--160, pp 531-543) COMPUTERS IN THE CORE SUPERVISION AND PROTECTION SYSTEM AT CREYS MALVILLE (SUPER PHENIX). Jeannot, A.; Grisolle, J.; Gourdon, J.; Paziaud, A. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France)). Jul 1976. (In French).

From IAEA/NPPCI specialists meeting on use of computers for protection systems and automatic control; Neuherberg/Munchen, F.R. Germany (11 May 1976).

In French nuclear power plants there is a tendency to distinguish between conventional handling of information by a computer (monitoring of a measurement with respect to a threshold, contact monitoring) and management of the process according to each reactor type. In case of the Super Phenix the process management has been voluntarily subdivided by specialized tasks in order to increase the availability and in order to facilitate the intervention into the logic part during start-up and operation. The interconnection of 5 computers lead to numerous precautions to avoid that they might disturb each other and to meet the current regulations.

- 2305 (MRR--160, pp 573-584) DESIGN FEATURES OF THE FUEL ELEMENT COMPUTERIZED PROTECTION SYSTEM BESSY. Juengst, U. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). Jul 1976.

From IAEA/NPPCI specialists meeting on use of computers for protection systems and automatic control; Neuherberg/Munchen, F.R. Germany (11 May 1976).

In the LMFB SNR 300 a process computer will be applied as a subsystem of the reactor protection system. It is to detect plugging fuel elements by monitoring the outlet temperatures of the 205 subassemblies. The system consists of four computers, three with 2 out of 3 majority voting for the task of engineered safeguards and one for the information output. The presentation describes the structural design of hard- and software and discusses some safety related features which may be considered by further integration of computers in reactor protection systems.

- 2306 (MTR--7485(Vol.1)) SYSTEMS DESCRIPTIONS AND ENGINEERING COSTS FOR SOLAR-RELATED TECHNOLOGIES. VOLUME I. SUMMARY. (Mitre Corp., McLean, VA (USA). METREK Div.). Jun 1977. Contract EX-76-C-01-2322. 157p. Dep. NTIS, PC A08/MF A01.

The results of the METREK solar energy engineering cost study, in support of the System to Project Utilization of Renewable Resources (SPURR) model, are presented. Generic systems designs are formulated and the associated engineering costs are evaluated and estimated. An aim of this study is to predict the likely costs and performance of solar equipment, once mass produced. The data have been summarized on generic design data forms and are presented. The systems are grouped into four major categories by application type: (1)

electric utility applications, (2) solar heating and cooling of buildings (SHADESIGN, nment contractors DB), (3) agricultural and industrial process heat (A/IPH), and (4) synthetic fuels and products (SFP). The electric utility application data are presented for five classes of systems: baseload, intermediate, semi-peaking, peaking and 'fuel savers'. Fuel savers are solar electric options without storage or backup. The SHACOB data are presented for five system types covering a wide range in demand load. These systems include domestic hot water only (liquid); heating and hot water (air and liquid); heating, cooling by absorption air-conditioning, and hot water (liquid); and heating, cooling and hot water with heat pump auxiliary (liquid). The A/IPH data are presented for eight applications using water, air and steam at delivery temperatures ranging from 40 to 800°C. System performance is estimated for nine locations representative of the census regions in climate, with proximity to industrial centers of concentration. The SFP data are delineated by product type. The only product applications considered are low and medium Btu gas, synthetic (high Btu) natural gas (SNG), synthetic crude oil (syncrude), methanol and ammonia. Costs for SFP systems are developed for biomass feedstocks, high sulfur coal, oil shale, and an equatorially-located ocean thermal energy conversion (OTEC) system.

- 2307 (MTR--7485(Vol.4)) SYSTEMS DESCRIPTIONS AND ENGINEERING COSTS FOR SOLAR-RELATED TECHNOLOGIES. VOLUME IV. CONVENTIONAL TECHNOLOGIES FOR ELECTRIC UTILITY APPLICATIONS. Miller, G. (Mitre Corp., McLean, VA (USA). METREK Div.). Jun 1977. Contract EX-76-C-01-2322. 101p. Dep. NTIS, PC A06/MF A01.

This document presents engineering costs and system performance for competitive fossil- and nuclear-fueled generation technologies. The data are developed to provide a data base for predicting future market penetration of solar technologies. To provide an accurate and consistent basis for market penetration analyses, the work reported in this volume was based on a set of assumptions common to all volumes and use of methodology consistent with that used for solar technologies. Competitive fossil- and nuclear-fueled generation technologies include those technologies expected to be most competitive between 1980 and 2020. On the basis of choosing a wide variety of technology types and of choosing those technologies exhibiting relatively competitive costs, eight technologies were chosen and are discussed: conventional coal/steam with flue gas desulfurization; open-cycle gas turbine/steam turbine with integrated coal gasification; atmospheric fluidized bed; light water reactor; liquid metal fast breeder reactor; open-cycle gas turbine with recuperator; open-cycle gas turbine-combined cycle; liquid metal fast breeder reactor and coal steam plant incorporating thermal oil storage. Most of the technologies are expected to have demonstrations completed by 1990. The LMFB and thermal oil storage are expected to have been demonstrated shortly after 1990. Generally the dates for demonstration represent optimistic estimates of the earliest timeframe within which prototype units could be completed. Sections 8 and 9 discuss the availability and anticipated costs of coal and nuclear fuels, respectively. (MCW)

- 2308 (N--78-11243) EFFECT OF SIMULATED SERVICE CONDITIONS ON STAINLESS STEEL TYPE 316. Kent, J.C.W.; Infield, J.A. (Bristol Univ.

(UK). Dept. of Mechanical Engineering). Jun 1977. 52p. (BU--77/B.4). NTIS PC A04/MF A01.

Materials used in high temperature applications are often subject to a departure from their virgin properties due to severe service conditions. It is therefore necessary to investigate the changes in the microstructure and mechanical properties of materials used in such applications in order to avoid premature and costly failure. The project was devised in order to determine the effect of prior damage and aging on stainless steel type 316 which is used in the construction of the fast breeder nuclear reactor. This was accomplished by subjecting prestrained, fatigued, and cold worked specimens to an accelerated aging process and observing subsequent changes in the tensile, hardness, creep, and impact properties of the material. The results obtained showed that for annealed material, irrespective of damage, the tensile strength increased with aging time after an initial period of weakening. Prior fatigue and prior strain were observed to accelerate these effects. Cold worked material tended to deteriorate continuously with aging, while the impact and hardness properties of all pretreated types of alloy decreased with aging.

2309 (NCSR-R--20, pp 89-117) INITIAL DATA COLLECTION EFFORTS OF CREDO. SODIUM VALVE FAILURES. Bott, T.F.; Haas, P.M. (Oak Ridge National Lab., TN (USA)). Nov 1978. From 5. symposium on reliability technology; Bradford, UK (Sep 1978).

The Centralized Reliability Data Organisation (CREDO) has been established at Oak Ridge National Laboratory to define, develop, and maintain a reliability data analysis center for use in advanced reactor safety and licensing. Its primary functions are collection, reduction, evaluation, storage, retrieval, and dissemination of reliability/maintainability data. Data-collection efforts have been initiated at several test loops, at the Experimental Breeder Reactor-II and at the Fast Flux Test Facility. Top priority is being given to collection data on safety and safety-related systems, primarily for sodium-cooled reactors. Sufficient operating time has been accumulated on sodium valves at test facilities to provide quantitative estimates of reliability characteristics with a reasonable degree of confidence. Sodium-valve failures have been categorized according to seat design, size, seal type, and actuator type.

2310 (ND-R--9(S)) COMPARISON OF THE APPLICATION OF PLATE AND BEAM THEORIES TO THE ELASTIC DILATION AND INTERACTION OF PFR SUB-ASSEMBLY WRAPPERS. Moss, R.L. (UKAEA Springfields Nuclear Power Development Labs.). 23 Jun 1977. 26p. Dep. NTIS (US Sales Only), PC A03/MF A01.

A wrapper face is assumed to be a long, narrow, rectangular plate. The mechanical interaction between adjacent dilating wrappers in contact along an axial line is discussed in terms of the theory of the bending of plates. A variational method is used to obtain neat and concise equations that determine both the interaction load and the length of the line of contact. The prime objective of the work is to compare the results obtained from plate theory with corresponding expressions from much simpler calculations based on beam theory. Numerical results indicate that the elastic dilation of a wrapper and its interaction with a neighboring wrapper can be calculated to adequate accuracy by simple beam theory.

2311 (ND-R--50(R)) SPECIFICATION OF SYSTEM - INDEPENDENT SEQUENCING CALLS FOR MODULAR CODE SCHEMES. Enderby, J.A. (UKAEA Reactor Group, Risley). Oct 1977. 14p. Dep. NTIS (US Sales Only), PC A02/MF A01.

To permit the writing of program modules which may be used without modification in any modular code scheme, a set of FORTRAN subroutine calls for setting up and communicating with modular code sequences has been defined. These calls, which are presented in the paper, have been defined so as to be independent of the way in which sequencing is implemented by the modular code scheme.

2312 (ND-R--178(R)) WHOLE-CORE ACCIDENTS IN AN LMFBR. Baker, A.R. (UKAEA Risley Nuclear Power Development Establishment). Nov 1978. 28p. Dep. NTIS (US Sales Only), PC A03/MF A01.

The lecture presents an introduction to the evaluation of whole-core accidents in a liquid-metal cooled fast breeder reactor (LMFBR). These accidents can be sub-divided into two types: transient overpower accidents or transient under-cooling accidents, according as the power or coolant flow is substantially disturbed from normal values. There would only be serious consequences in the event of failure of the reactor's automatic protection system or, for a few less important hypothetical accidents, in the event of sudden and catastrophic failure of a major structure. The treatment of whole-core accidents in safety presentations to responsible authorities is briefly discussed and the possible courses of such accidents are outlined, showing the importance of the Doppler fuel temperature and sodium void coefficients. The calculational methods are described with illustrations from studies with the UK code FRAX. The lecture concludes with brief consideration of the experimental validation of computer codes and the design consequences of whole-core accident studies.

2313 (ND-R--199(R)) VIBRATION TESTS ON SINGLE HEAT EXCHANGER TUBES IN AIR AND STATIC WATER. Collinson, A.E.; Warneford, I.P. (UKAEA Springfields Nuclear Power Development Labs.). Jul 1978. 25p. Dep. NTIS (US Sales Only), PC A03/MF A01.

The vibrational characteristics of a 7 span straight tube and a 26 span U-tube have been investigated for the effects of fluid medium (air/water), tube-grid clearance, tube-grid contact force, vibration transmission and scale. Measured frequency response and mode shapes compared favourably with theoretical values, vibration with pin-pin tube support being most readily excited. The frequency reduction on immersion in water corresponded to an added mass equivalent to the liquid displaced mass. Dynamic magnifiers varied in the range 12 to 135 with mean values of 30 to 40 in water and 45 to 60 in air. Principal vibration modes and damping values were reproducible in a half-scale model of a U-tube.

2314 (ND-R--230(R)) NOISE PROBLEMS IN PFR SAFETY SYSTEM POWER SUPPLIES. Hodgson, D. (UKAEA Risley Nuclear Power Development Establishment). Aug 1978. 10p. (ND-M-R--230(R)). NTIS, PC A02/MF A01.

Reactor power on PFR is controlled by eleven neutron absorbing rods. They are held vertically above the core by electromagnets which, in turn, are mounted on motorised lead screws. Five of the rods are used as power control elements; five are used as shut off

rods and the eleventh is a safety rod, positioned vertically above the centre of the reactor core. All rods fall under gravity into the reactor when the electro magnets are de-energised. In addition the safety rod can be driven into the reactor by a pneumatic piston and cylinder mechanism. This provides some degree of diversity from the common design of mechanism used by the control and shut off rods. Also being in the center it can be armed and ready for use when the above core shield is rotated. The automatic protective system for the plant provides the electro magnet currents and hence the reactor trip signals via two separate and diverse safety systems, namely a 'Relay System' and a 'Solid State System.' The two systems are completely separate and independent. The overall probability of failure of the complete system to release at least 3 rods when called upon to do so is less than 10^{-6} .

- 2315 (ND-R--304(R)) INTERNATIONAL PIPING BENCHMARKS: USE OF THE SIMPLIFIED CODE PACE 2. Boyle, J.T.; Spence, J. (UKAEA Risley Nuclear Power Development Establishment). May 1979. 31p. Dep. NTIS, PC A03/MF A01.

Work performed under United States-UKAEA Fast Reactor Exchange Programs.

This report compares the results obtained using the code PACE 2 with the International Working Group on Fast Reactors (IWGFR) International Piping Benchmark solutions. PACE 2 is designed to analyse systems of pipework using a simplified method which is economical of computer time and hence inexpensive. This low cost is not achieved without some loss of accuracy in the solution, but for most parts of a system this inaccuracy is acceptable and those sections of particular importance may be reanalysed using more precise methods in order to produce a satisfactory analysis of the complete system at reasonable cost.

- 2316 (NEDM--13981) STATE-OF-THE-ART OF RUPTURE DISKS FOR LMFBR APPLICATION. Verkamp, J.P. (General Electric Co., Sunnyvale, CA (USA). Breeder Reactor Dept.). Jul 1973. Contract EY-76-C-03-0189-061. 73p. AT.

Portions of document are illegible.

The Liquid Metal Fast Breeder Reactor (LMFBR) power plants as presently conceived have sodium heated steam generators. Steam-side-to-sodium-side leakage is possible. It is expected that safe relief of the products generated by a sodium/water reaction can be obtained by use of rupture disks. In this review the present state of the technology is examined with regard to rupture assemblies suited to LMFBR applications. Requirements to meet LMFBR needs are compared with characteristics of presently available rupture assemblies. Recommendations for meeting near term and future needs are included.

- 2317 (NEDM--14044) SMALL LEAK DAMAGE AND BEHAVIOR TEST RESULTS. Greene, D.A. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Apr 1975. Contract EY-76-C-03-0893-010. 33p. AT.

Portions of document are illegible.

Small leak damage and behavior tests have been performed in a static sodium rig, a flowing sodium rig, and a rig with flowing sodium and flowing steam simulating LMFBR Steam Generator conditions. Approximately 160 wastage behavior tests have been completed in these rigs. The results obtained from these tests are presented. They include target damage rates, and the magnitude of orifice self-damage.

- 2318 (NEDM--14090) SODIUM WATER REACTION (SWR) CLEANUP SYSTEM STUDY AND IMPROVED DESIGNS OF COLD TRAP FOR SWR CLEANUP OF LMFBR PLANTS. Anand, R.K.; Reynolds, R.G. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jan 1976. Contract EY-76-C-03-0893-010. 95p. AT.

Portions of document are illegible.

This report documents an SWR Cleanup System Study for the hot sodium circulation process of removing sodium water reaction products (SWRP) from an Intermediate Heat Transport System (IHTS) of an LMFBR power plant. In essence, the SWR Cleanup System Study concludes that there will be a saving of several million dollars if a special fast SWR Cleanup System proposed in this report is used for in place removal of SWRP. A survey of the current state of knowledge regarding cold trap design and operation is presented. Two improved design concepts are recommended for SWR cleanup cold traps which incorporate the good design principles required for cleanup of highly impure sodium. Development programs are also recommended for a successful demonstration of the recommended sodium cleaning process.

- 2319 (NEDM--14091) OVERALL SEPARATION EFFICIENCY REQUIREMENT OF THE CENTRIFUGAL SEPARATOR FOR THE LMFBR SODIUM WATER REACTION PRESSURE RELIEF SYSTEM (SWRPS). Anand, R.K. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jan 1978. Contract EY-76-C-03-0893-010. 21p. AT.

An analysis was made to determine the overall separation efficiency requirement for a centrifugal separator in a LMFBR plant to stay within recommended Na_2O ground deposit limits and NaOH -air concentrations. Expected release from hypothetical sodium-water reactions associated with LMFBR steam generator is discussed in terms of effluent generation, atmospheric diffusion, release height and subsequent ground deposition. The analysis concludes that a centrifugal separator efficiency of 94 percent or greater will maintain NaOH concentrations well below the recommended allowable limits necessary to protect the health and safety of the public.

- 2320 (NEDM--14092) SMALL LEAK SHUTDOWN, LOCATION, AND BEHAVIOR. Sandusky, D.W. (General Electric Co., San Jose, CA (USA). Fast Breeder Reactor Dept.). Dec 1975. Contract EY-76-C-03-0893-010. 36p. AT.

The work reported is a continuation of an investigation of the nature and control of small steam leaks in LMFBR steam generators. The intent of the present work was to test methods to shut down a leaking steam generator and develop methods to locate small leaks. The testing was performed in a system designed to simulate steam and sodium conditions in an LMFBR steam generator (Loop 9). Shutdown tests were first performed to determine if any method would keep a small leak open. High pressure inert gas proved successful. Procedures involving cooling the sodium to $<316^\circ\text{C}$ before replacement of steam with inert gas proved successful. Cooling the sodium reduced the small leak self-damage rate. A preliminary shutdown procedure was determined.

- 2321 (NEDM--14104) THEORETICAL PREDICTIONS OF LOCAL HEAT TRANSFER AND WATER PURITY PARAMETERS FOR TRANSITION BOILING BASED UPON A FILM DRYOUT MODEL. Peterson, J.R. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Mar 1976. Contract EY-76-C-03-0893-010. 54p. AT.

The report presents the development of a theoretical prediction of local heat transfer coefficients, wall temperatures and lengths for the Transition Region between the nucleate boiling and film boiling areas of an evaporator tube. The theoretical expressions are based upon a physical model which envisions the annular liquid film, thought characteristic of nucleate boiling at the conditions of interest, to break up into rivulets at the critical heat flux point. The length of the Transition Region is assumed to be dominated by the evaporation of these rivulets, and the dryout point for the rivulets is equated to the onset of film boiling. The model is solved for controlled heat flux and controlled temperature boundary conditions as well as for the liquid metal heated boundary condition appropriate for LMFBR steam generators.

- 2322 (NEDM--14109) SUMMARY OF THE EFFECTS OF MATERIAL COMBINATIONS AND WELD JOINT GEOMETRIES ON THE ELASTIC STRESS IN A DISSIMILAR METAL PIPE JOINT. Yang, T.M. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). May 1976. Contract EY-76-C-03-0893-010. 21p. AT.

An investigation of the temperature and thermal stress distribution in a 24 inch diameter dissimilar metal pipe joint during a thermal transient has been completed. Two material combinations, $2\frac{1}{4}\text{Cr}-1\text{Mo}$ to 316SS and $2\frac{1}{4}\text{Cr}-1\text{Mo}$ to Incoloy 800 were studied to determine the effect of material combination and weld joint geometry on thermal stresses. The study showed that material combination had a large effect on stresses; the smaller difference in thermal expansion resulted in lower stresses. Weld geometry had a secondary effect on thermal stresses.

- 2323 (NEDM--14110) CAUSTIC STRESS CORROSION TESTS FOR THE LLTR. Indig, M.E. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). May 1976. Contract EY-76-C-03-0893-010. 30p. Dep. NTIS, PC A03/MF A01.

A series of tests have been performed in order to determine the effects of the caustic resulting from the $\text{Na}/\text{H}_2\text{O}$ reaction on the materials used in the LLTR-MSG series of testing. Stainless steel, $2\frac{1}{4}\text{Cr}-1\text{Mo}$ and carbon steel have been evaluated. Stress corrosion cracking susceptibility and general corrosion are reported. Over the range of temperature, caustic concentration and heating rate tested the stainless steel stressed to 90% of yield or above suffered cracking. Whereas, the $2\frac{1}{4}\text{Cr}-1\text{Mo}$ and carbon steel were not cracked.

- 2324 (NEDM--14131) RESULTS OF SMALL LEAK BEHAVIOR AND DAMAGE PROPAGATION TESTS IN THE U.S. Greene, D.A.; Sandusky, D.W. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Aug 1976. Contract EY-76-C-03-0893-010. 34p. AT.

A small leak of water into sodium in a liquid metal heated steam generator can cause damage to adjacent tubes, a phenomenon termed wastage. Theories on this phenomenon range from corrosion from sodium-water reaction products to erosion by supersonic particles. An alternative approach considers the water injection to form a simple combustion process. Using this approach many aspects of over 250 wastage experiments can be explained both analytically and physically. It is also found that water leaking into sodium causes self-enlargement of the defect, a phenomenon termed self wastage. Tests were performed in flowing and static sodium and steam. Self-wastage

appears to result from two separate causes; the first operating on plugged leaks possibly causing intermittent opening of the leak path, the second operating when a continuous flow of water passes through the leak path. Data on self-wastage were obtained for both austenitic and ferritic steels. When a small leak occurs and is detected, the steam generator shutdown sequence must be optimized to prevent intrusion of sodium into the steam system or plugging of the leak. Tests were performed using various shutdown procedures to identify optimum conditions for leak location. Methods to reopen plugged leaks were also tested. Some of the shutdown procedures repeatedly produced open leaks which were subsequently located with a helium mass spectrometer. Washing with chemical solutions successfully opened plugged leaks. It was found that pulsed eddy current and ultrasonic inspection appear promising for leak location techniques. Damage propagation and subsequent failure causes are being examined, and predictions made using combustion flame approaches. The flame characteristics have been measured, and the damage potential of both turbulent and pool flames have been examined.

- 2325 (NEDM--14137) SMALL LEAK SCOPING TESTS: SECONDARY TUBE FAILURE TEST; SODIUM/WATER REACTION FLAME PHOTOGRAPHY. Gudahl, J.A.; Hunsicker, J.C.; Ulpindo, C.Q. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Aug 1976. Contract EY-76-C-0893-010. 36p. AT.

Two Sodium/Water Reaction Scoping Tests are described. In the first test, a 19 tube bundle of $2\frac{1}{4}\text{Cr}-1\text{Mo}$ material was assembled in a 1.22 inch (3.1 cm) triangular array, simulating the Clinch River Breeder Reactor Plant steam generator. The assembled test section was placed in a liquid sodium test vessel and a steam jet was directed through the liquid sodium to a target tube. The initial injection of water was directed through the space between two tubes to impinge on a tube 1.5 inches (3.9 cm) from the injection orifice. This target tube was pressurized with steam. The impinging jet penetrated the 0.109 inch (.8 cm) target tube after 20 minutes. Steam from the target tube then impinged onto one of the adjacent tubes causing it to fail after a further 63 seconds. The test and test results are described. The second group of tests was conducted to obtain photographs of a liquid sodium/water reaction. Combustion occurred in all six tests and in most cases, visual documentation, color still and cinema photography, showed that a flame or flame pattern was produced.

- 2326 (NEDM--14145) CHARACTERIZATION OF ESR AND VAR $2\frac{1}{4}\text{Cr}-1\text{Mo}$ ALLOY TUBING. Bubsom, H.J. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1976. Contract EY-76-C-03-0893-010. 30p. Dep. NTIS, PC A03/MF A01.

Commercial tubing for LMFBR sodium-water steam generator applications was produced from ESR and VAR $2\frac{1}{4}\text{Cr}-1\text{Mo}$ alloy (Grade T22) and characterized regarding chemical composition, microstructure, physical characteristics, and short and long term mechanical properties. These results demonstrate that tubing meeting RDT Standard RDT M3-33 can be produced from either the ESR or VAR process and is acceptable to ASME Code Section III requirements. Metallurgical and mechanical properties are similar for both ESR and VAR material, indicating that either remelting practice is acceptable.

- 2327 (NEDM--14146(Rev.1)) EFFECTS OF REMELTING AND HEAT TREATMENT ON 2-1/4 CR-1MO STEEL ELEVATED TEMPERATURE TENSILE PROPERTIES. Offer, H.P.; Copeland, J.F.; Yuen, J.L. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1976. Contract EY-76-C-03-0893-010. 70p. Dep. NTIS, PC A04/MF A01.

Vacuum arc remelting (VAR) and electroslag remelting (ESR) processes for steam generator tubesheet forgings and tubing are to be used at the CRBR. It was found that there is essentially no significant effect on the uniaxial tensile properties of 2 1/4 Cr-1Mo steel by VAR or ESR. A summary is presented of results of a study of heat treatment combined with post weld heat treatment. It was found that regardless of the primary heat treatment, the post weld heat treatment plus in-service conditions will reduce the yield strength at elevated temperature below the Code Case expected minimum curve. (FS)

- 2328 (NEDM--14149) INVESTIGATIONS INTO THE SODIUM-STEAM SURFACE INTERACTION ON A STAGNANT POOL. Stehling, S.G.; Hunsicker, J.C. (General Electric Co., Sunnyvale, CA (USA). Breeder Reactor Dept.). Sep 1976. Contract EY-76-C-03-0893-010. 53p. AT.

Experimental investigations into the interface reaction between a steam atmosphere and stagnant sodium pool have been conducted at the General Electric San Jose Test Facility. The objectives of the tests were to create worst case accident situations and evaluate the consequences with respect to the design of LMFBR Steam Generators. The most important results of these investigations were the creation and actual photographing of a "pool fire" and the determination that the primary damage mechanism associated with it is of a corrosive nature rather than thermal weakening as previously suspected.

- 2329 (NEDM--14150) THERMAL/HYDRAULIC TEST RESULTS INCLUDING CRITICAL HEAT FLUX CONDITIONS FOR A SODIUM-HEATED STEAM GENERATOR TUBE MODEL. Wolf, S.; Holmes, D.H. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Sep 1976. Contract EY-76-C-03-0893-010. 56p. AT.

Thermal/hydraulic tests were performed with a sodium-heated water-steam tube prototypic of a CRBRP steam generator tube section. The test section was designed and is being used for DNB Effects testing to support engineering development of the CRBRP steam generators. The experimental ranges were: pressure, 1590 to 1890 psia (11.0 to 13.0 MPa); mass flux, 0.97 to 2.39 10^6 lbm/hr-ft² (4.74 to 11.7 10^6 kg/hr-m²); critical heat flux, 0.21 to 0.44 10^6 Btu/hr-ft² (0.66 to 1.39 MW/m²); and critical quality, 0.23 to 0.40. The thermal/hydraulic results of this test are analyzed and presented in this report. Specific results include: evaluation of critical steam quality correlations, determination of the length of the transition boiling region, and preliminary evaluation of tube wall thermal conductivity.

- 2330 (NEDM--14153) TEST RESULTS, LONG-TERM ENDURANCE TEST CONDITIONS, AND SENSITIVITY PARAMETERS FOR THE DNB EFFECTS TEST. Wolf, S.; Holmes, D.H. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). 8 Jun 1976. Contract EY-76-C-03-0893. 29p. AT.

Results of an evaluation of selected thermal/hydraulic data from the DNB Effects Test and analysis leading to a definition of thermal/hydraulic conditions for the long-term

endurance test to simulate the worst CHF condition occurring in CRBRP are reported. Analysis of data is based on THAN, a GE-developed data-reduction computer code. Specific results from THAN include a CHF quality correlation and a tube wall conductivity correlation for conditions corresponding to the long-term endurance test. The final long-term endurance test conditions are based on steam generator performance evaluation with the computer codes STMSYS and STMGEN utilizing the experimentally determined correlations from THAN. Sensitivity parameters for deviations from the nominal conditions corresponding to CHF location, Z/sub c/, and temperature oscillation, $\Delta(AT)/sub c/$, are also reported.

- 2331 (NEDM--14161) PRESSURE PULSE PROPAGATION IN PIPING ANALYTICAL/EXPERIMENTAL COMPARISON. Knight, D.D. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Dec 1976. Contract EY-76-C-03-0893-010. 65p. AT.

A comparison of analytical results to experimental data is given to illustrate the ability of the TRANSWRAP code to model the transient pressure pulse phenomena in a simple piping system including attenuation due to elbows and a tee. Elbow attenuation is produced within the TRANSWRAP code with user specified elbow loss coefficients. Parametric analyses were done to determine loss coefficients to produce results which best fit the data. The TRANSWRAP method of evaluating the effect of a tee with one-dimensional area change relationships was found to be accurate for a broad pulse (compared to pipe diameter) but inadequate for a narrow pulse.

- 2332 (NEDM--14164) EVALUATION OF STRAIN CYCLING EFFECTS IN THE DNB ZONE OF THE CRBR EVAPORATORS. Magee, P.M.; Casey, D.F.; Chus, C.L.; Dillman, C.W.; Roberts, J.M.; Wolf, S. (General Electric Co., Sunnyvale, CA (USA). Breeder Reactor Dept.). Dec 1976. Contract EY-76-C-03-0893-010. 156p. AT.

Departure from nucleate boiling (DNB) occurs in the water-steam tubes of the CRBR evaporators. The flow preceding the DNB (or dryout) point is dispersed annular flow characterized by a film of liquid water adjacent to the tube wall. Heat transfer is by nucleate boiling. As the liquid film evaporates, it decreases in thickness and, at DNB, the film finally breaks up into rivulets and dry spots. After the rivulets have evaporated the wall is covered by bulk vapor with entrained droplets, and heat transfer is by film boiling with a heat transfer coefficient considerably lower than for the wet region immediately preceding dryout. Axial oscillation and/or movement of the DNB point cause oscillations between wet-wall heat transfer coefficients, which in turn produce oscillations in the tube wall temperatures and thermally induced stresses which can significantly reduce tube life and, in the limit, lead to fatigue of the tube wall. In March 1976, a task force was established to evaluate the DNB design basis for the CRBR steam generator system (system 53). The basic objectives of the task force investigation were as follows: (1) Determine if a DNB thermal cycling problem actually exists in the CRBR evaporator and, if so, its magnitude; (2) determine if the current design approach (recirculation ratio constant at full power value) is feasible and provides 30 years DNB thermal cycle fatigue life; and (3) if the current design approach is not feasible, recommend necessary design changes, giving

strong consideration to cost and schedule impact on system 53. This report summarizes the task force's technical evaluations, analyses and recommendations regarding the system 53 design.

- 2333 (NEDO--12324) PU, U REDISTRIBUTION IN (U,Pu)O₂ FUELS BY TEMPERATURE GRADIENTS. Adamson, M.G.; Aitken, E.A. (General Electric Co., San Jose, CA (USA). Nuclear Energy Div.). Aug 1972. 16p. General Electric Co., Nuclear Energy Div., San Jose, CA.

Portions of document are illegible.

A predictive model of the time-dependent evolution of radial Pu/U + Pu gradients in operating nonmolten mixed oxide fuels is being developed. This model is based on preferential evaporation-condensation and vapor transport of metal-bearing species either along cracks and porosity channels within the fuel or inside closed pores migrating up the temperature gradient; equilibrium thermodynamics are applied to calculate the continually changing vapor composition over the hot fuel. Predictions of the model are used in combination with recent in-pile experimental data (from nonmolten mixed oxide fuel pins irradiated in EBR-II from 0.2 to about 11 at. % burnup at varying powers) to delineate the key parameters in radial actinide redistribution. A correlation between fuel stoichiometry (O-to-Pu ratio) and the degree of redistribution (described by an enrichment factor $\gamma_{\text{sub Pu}}$) identified, however, the degree of restructuring of the fuel - as determined by the peak linear power rating, time at this power, and the initial fuel density - also has a strong influence on the final value of $\gamma_{\text{sub Pu}}$. Certain kinetic features of the process and intended modifications to the present model are discussed, and evidence for axial actinide vapor transport in an irradiated annular fuel is also presented.

- 2334 (NEDO--12325) THERMODYNAMIC STUDIES OF THE SODIUM FUEL REACTION AT GENERAL ELECTRIC. Adamson, M.G.; Aitken, E.A.; Evans, S.K. (General Electric Co., Pleasanton, CA (USA). Vallecitos Nuclear Center). Aug 1972. 12p. General Electric Co., Vallecitos Nuclear Center, Pleasanton, CA.

Portions of document are illegible.

Available thermodynamic information for the equilibrium system sodium (l) + sodium-fuel compound + unreacted fuel is critically reviewed. Results are interpreted primarily in terms of the equilibrium oxygen concentration sodium (C_O) and the oxygen potential Δ anti G/sub O₂) of the remaining fuel phase. Results of series of isothermal and nonisothermal capsule tests performed at General Electric are presented and an alternative experimental approach utilizing a high temperature EMF cell is also described. Vanadium wires were used to determine C_O in the isothermal tests. Although there is significant scatter in the vanadium wire results, practically all the C_O values fall within the range 0.1 to 1 ppm (oxygen by weight) between 700 and 850°C. When expressed as oxygen potentials these data are considerably more negative than Δ anti G/sub O₂/ values obtained either from determination of the O/M ratio of the unreacted fuel phase or calculations based on the known free energy of formation of Na₂UO₄. Unusual kinetic features of the sodium-fuel reaction are suspected as a possible cause of this discrepancy, but there is also some doubt concerning the validity of the thermodynamic approximations used to describe dilute solutions of oxygen in sodium.

- 2335 (NEDO--12326) OXYGEN REDISTRIBUTION IN LMFBR FUELS. Evans, S.K.; Aitken, E.A. (General Electric Co., Pleasanton, CA (USA). Vallecitos Nuclear Center). Aug 1972. 5p. General Electric Co., Vallecitos Nuclear Center, Pleasanton, CA 94566.

The behavior of oxygen in hypostoichiometric mixed oxide fuels in a temperature gradient is discussed from experiments performed out-of-pile. Oxygen migration toward the cold end of the temperature gradient has been consistently observed. However, the degree of distribution, characterized by the heat of transport, is a function of the over all stoichiometry and the Pu content of the fuel. An empirical relation between the heat of transport and the over-all stoichiometry of typical LMFBR oxide fuels has been formulated and used to predict the O-to-M profile under a given set of irradiation conditions. Cladding oxidation imposes an additional constraint and must be properly accounted for in predicting O-to-M profiles. Other reactions involving fission products may influence the O-to-M profile under certain conditions.

- 2336 (NEDO--12668) CONSTANT EXTENSION RATE TESTING FOR LMFBR STEAM GENERATOR APPLICATIONS. Indig, M.E. (General Electric Co., San Jose, CA (USA). Nuclear Energy Div.). May 1977. Contract EY-76-C-03-0893-027. 39p. (CONF-770524--4). Dep. NTIS, MF A01.

From ASTM symposium on fatigue testing of weldments; Toronto, Canada (May 1977).

Portions of document are illegible.

A high-temperature constant extension rate facility was developed to evaluate the stress corrosion resistance of alloys considered for the construction of LMFBR steam generators. The primary concern was stress corrosion that might occur on the water side during steam generator operation in locally faulted caustic environments. Most of the studies were performed at 316°C in 5 or 10% NaOH solutions with 2-1/4 Cr-1 Mo steel and Incoloy-800. With minor modification, the constant extension rate test was converted to a straining electrode test. Experiments conducted by the straining electrode test permitted the study of stress corrosion behavior in the high-temperature caustic solutions over a range of oxidizing potentials. It was found that 2-1/4 Cr-1 Mo steel was extremely resistant to caustic cracking at 316°C in all metallurgical conditions tested by the constant extension rate test and the straining electrode test. A few constant extension rate tests were performed at 232°C in 5 and 10% NaOH. At the lower temperature 2-1/4 Cr-1 Mo failed by intergranular and transgranular stress corrosion. Unlike 2-1/4 Cr-1 Mo, Incoloy-800 was susceptible to caustic cracking at 316°C. The susceptibility depended on the metallurgical condition and to some extent, the oxidizing potential. The cold-worked and Grade 1 conditions exhibited excellent resistance to caustic cracking in the constant extension rate test. However, limited studies in the straining electrode test showed the Grade 1 condition could be readily cracked at anodic oxidizing potentials.

- 2337 (NIIAR--15(309)) NEUTRON SPECTRUM MEASUREMENT IN THE BDR-60. Nigmatullin, N.R.; Gadzhiev, G.I. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov, Dimitrograd (USSR)). 1977. 15p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

The results of neutron spectrum measurement at the various points of the core radius, of the side shield and of the MOP-60 reactor vertical experimental channel by the foil

activation method have been given. The used technique is based on a regularization method, developed to solve incorrect problems and permits to restore spectra within the range from 0 to 20 MeV. Maximum neutron flux at the reactor centre is 5.05×10^{13} and in the vertical experimental channel is 6.7×10^{11} neutr./($\text{cm}^2 \cdot \text{cmw}$). Spectrum experimental data for the core centre (up to R approximately 13 cm) are in a good agreement with the calculation data. Spectrum experimental data for the area change insignificantly, and the integral current density decreases approximately 1.2 time. The most divergence of the experimental and calculation data is observed on the core boundary and the side shield, where the share of neutrons with the energy less than 10 keV decreases approximately 4 times as compared to the calculation.

- 2338 (NIIAR-P--2(296)) RADIOACTIVE IMPURITIES TRAPPING BY THE BOR-60 COLD TRAP. Kizin, V.D.; Lisitsyn, E.S.; Poznyak, G.I.; Polyakov, V.I.; Chechetkin, Yu.V. (Nauchno-Issledovatel'skiy Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). 1977. 21p. (In Russian). Dep. NTIS (US Sales Only), PC A03/MF A01.

Experimental studies were made to elucidate the accumulation of radioactivity in the standard cold trap (120 deg C) of the BOR-60 reactor. The most dangerous isotopes (possessing maximum specific radioactivity, long half-life period, gamma-quantum energy in excess of 600 keV) in the circuit of the fast reactor incorporating defective fuel elements are represented by isotopes of ^{137}Cs and ^{134}Cs . It is shown that in the cold trap the following isotopes are concentrated: ^{65}Zn , ^{134}Cs , ^{136}Cs , ^{137}Cs , ^{131}I , ^{60}Co , ^{124}Sb , and ^{22}Na . It should be noted that sodium in the trap is almost completely cleaned of iodine and tellurium, while the radioactivity of cesium, antimony and cobalt isotopes in the trap constitutes 20 - 30% of their content in the first circuit. It is established that the ability of the trap to catch cesium is not directly associated with its mutual precipitation with the oxides, sorption on the oxides or filtration. A conclusion is made that the cold traps do not represent an effective means of reducing the radioactivity level of the sodium coolant.

- 2339 (NIIAR-P--4(270)) TWO-DIMENSIONAL MODEL OF A LARGE POWER SODIUM-COOLED FAST BREEDER REACTOR WITH AN OXIDE FUEL FOR TEST CALCULATIONS. Zizin, M.N.; Kudryashov, L.N.; Nikolaev, M.N. (Nauchno-Issledovatel'skiy Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). 1976. 33p. (In Russian). Dep. NTIS (US Sales Only), PC A03/MF A01.

The suggested two-dimensional model of a fast breeding reactor is intended for performing comparative test calculations aimed at the comparison between the reactor characteristics calculated with the aid of the programmes and the constants employed in various research centres. The model corresponds to the reactor of the nuclear power plant with an electric power of 1500 MW. The technique of employing the model for testing is presented. The system of 6-group constants is proposed and the method of dividing the reactor volume by means of the end-difference grid for the basic calculations is shown. The initial data for the basic calculations has been the system of 26-group constants with regard to the changes in the group cross-sections of the main fissionable isotopes.

- 2340 (NIIAR-P--4(338)) REACTIVITY EFFECTS

ANALYSIS AS A MEANS OF FAST REACTOR DIAGNOSTICS. Afanas'ev, V.A.; Gryazev, V.M.; Efimov, V.N.; Krasnoyarsk, N.V.; Skulkin, N.N. (Nauchno-Issledovatel'skiy Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). 1978. 24p. (In Russian). Dep. NTIS (US Sales Only), PC A03/MF A01.

The components of the temperature and power reactivity coefficients of the BOR-60 reactor have been experimentally and theoretically investigated. The dependence of the coefficients on the reactor power, fuel burnup, coolant flow rate and inlet temperature have been measured. In measuring the coefficients by a dynamic method the time constants of some reactor constituents have been determined. A satisfactory agreement has been observed between theory and practice. The temperature coefficient has proved weakly dependent on the reactor parameters and time. The power coefficient has a complicated dependence on the parameters and decreases with time. The developed technique of calculating the reactivity balance alongside the analysis of the reactivity components provides for early diagnostics of the reactor.

- 2341 (NIIAR-P--5(271)) STUDY OF BOR-60 REACTOR FUEL ELEMENT PERFORMANCE. Tsykanov, V.A.; Davydov, E.F.; Gryazev, V.M. (Nauchno-Issledovatel'skiy Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). 1976. 20p. (In Russian). Dep. NTIS (US Sales Only), PC A03/MF A01.

Described are the test conditions and the results of the material researches of a series of the fuel element packets irradiated in the BOR-60 reactor up to burning-out of 6 to 11% of heavy atoms. Considered are the states of the jackets made of the OKh16N15M3B and 1Kh18N10T steels and fuel in the form of sintered uranium dioxide sleeves, effect of the irradiation parameters on the mechanical properties of the jacket, variation of the fuel element diameter and jacket density, accidents involving the breakdown of the fuel element jackets. It is shown that variation of the jacket diameter is due mainly to steel swelling. After irradiation up to burning-out of 8 - 11% of heavy atoms, the plasticity of the jackets has decreased by several tenth fractions of percent. It has been revealed that corrosion interaction of the fuel and jacket extends to the depth of up to 70 μm . It is concluded that the sealed fuel elements remain serviceable up to burning-out of 10 - 11% of heavy atoms.

- 2342 (NIIAR-P--5(299)) RADIOACTIVITY OF LONG-LIVED NUCLIDES IN THE BOR-60 PRIMARY CIRCUIT DURING OPERATION WITH SOME DEFECTIVE FUEL ELEMENTS. Gryazev, V.M.; Kizin, V.D.; Lisitsyn, E.S.; Polyakov, V.I.; Chechetkin, Yu.V. (Nauchno-Issledovatel'skiy Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). 1977. 17p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

Presented are the generalized results of measurement of the accumulation and distribution of the corrosion radioactive nuclides and fission products, which have been gained during four years of service of the BOR-60 reactor, including long-time operation of the defective fuel elements in the active zone. It is shown that when the quantity of the defective fuel elements amounts to about 1%, the radioactive situation becomes worse mainly owing to the escape of cesium radioisotopes and their accumulation in the coolant. As to other fission products, the maximum contribution to the dose rate of the pipelines and equipment is made by ^{140}Ba , ^{140}La and ^{95}Nb . When the quantity of the defective fuel elements equals

0.1-0.2%, this contribution is identical to the contribution made by the corrosion products of ^{60}Co and ^{54}Mn . The radioactivity of the corrosion products in the circuit has not increased within the last three years and has been by one order of magnitude below the calculated value. The corrosion and fission products are non-uniformly distributed in the circuit. The concentration of ^{95}Nb and ^{60}Co in "cold" sodium pipeline was by 2-5 times more, and concentration of ^{140}Ba and ^{54}Mn was by 10-20 times more, than in "hot" sodium pipelines. It has been established that cobalt is deposited basically in the heat exchanger. The efficiency of coolant drain from the pipeline is estimated with a view to reducing the dose rate.

- 2343 (NIIAR-P--17(283)) DECLADDING OF BOR-60 AND BR-5 FUEL ELEMENTS BY CLAD MELTING. Barkov, V.A.; Bibikov, S.E.; Vasil'ev, G.I.; Ivanov, P.M.; Novoselov, G.P.; Pakhomov, V.V.; Serov, A.V.; Skiba, O.V. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). 1976. 16p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

Experiments have been carried out on thermal decladding of irradiated oxide fuel elements of BOR-60 reactors (maximum burning 11%, exposure 6 and 3 months) and BR reactors (burning up to 5%, exposure 13 years). Uranium yield during decladding has been 99.9%, that of plutonium - 98.8%, steel concentration in the opened fuel - 5% and less. The physical condition of the fuel after the fuel element casing has been removed does not show any variations. The oxide fuel is shown to give up to 80% cesium, and partially iodine on heating up to 1870°K, while other splinter elements evolve in insignificant amounts.

- 2344 (NIIAR-P--18(284)) EXPERIMENTAL REPROCESSING OF IRRADIATED URANIUM FUEL FROM BOR-60 REACTOR BY FLUORIDE VOLATILITY METHOD. Kikoin, I.K.; Tsykanov, V.A.; Dem'yanovich, M.A. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). 1976. 13p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

The results are described of experiments on fluoride regeneration of irradiated assemblies of the BOR-60 reactor with uranium dioxide, with an average burn-up about 10% and a hold-up time of 3 and 6 months prior to reprocessing. In the course of the experiments, about 4300 g of fuel of the "hot" portion of the assemblies has been regenerated with a 90% initial enrichment with respect to uranium-235, and 4738 g purified uranium hexafluoride obtained, which has been reprocessed into a crystalline dioxide. Using the regenerated fuel and applying the vibrocompaction method, fuel rods have been prepared for re-use in the reactor. The distribution of fission products and the total radioactivity all over the experimental process chain equipment and the regeneration products has been investigated. Data on the yield and purification of uranium, the amount and activity of the process waste are presented.

- 2345 (NIIAR-P--18(312)) DETERMINATION OF BOR-60 REACTOR DYNAMIC CHARACTERISTICS USING DATA FROM STATISTIC EXPERIMENTS. Adamovskii, L.A.; Kebabze, B.V. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). 1977. 18p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

Considered are the problems associated with

performance of a passive statistical experiment during investigation of the dynamic characteristics of an object. The procedure and equipment of the experiments conducted on the BOR-60 reactor are described. Presented is the procedure for calculating the pulse characteristic by reference to the experimental estimation of the correlation functions with the use of the regularization method. The efficiency of various types of regulators is described, and practical recommendations are given. Described are the results of measurements of the consumption - neutron flux in channels and those of the consumption - coolant temperature at the outlet of the reactor core, which have been obtained with the use of the standard transducers of the BOR-60 reactor measurement system. Analyzed is the effect of the non-correlated disturbance in the registered signal noise on the accuracy of measurements. Discussed are the problems of increasing the statistical accuracy of measurements and improving the procedure for recording and processing of the noises.

- 2346 (NIIAR-P--24(358)) SODIUM BOILING EXPERIMENTS AT THE BOR-60 REACTOR. Afanas'ev, V.A.; Aleksandrov, K.A.; Kebabze, B.V. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). 1978. 11p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

Experimental results on the diagnostics of sodium boiling in the BOR-60 reactor core are presented. Detailed information is given on the boiling fuel assembly and the experimental set-up. Sodium boiling in the assembly was achieved by gamma-heating of tungsten rods. The feasibility of boiling detection by acoustic emission and neutron noise is shown. Some results of spectral and correlation analyses are provided. The need for special signal shaping is stressed to improve the detection reliability.

- 2347 (NIIAR-P--27(321)) COMPUTERIZED SUBSYSTEM FOR CONDUCTING INVESTIGATIONS AT THE BOR-60 REACTOR. Efimov, V.N.; Kachalin, V.A.; Sviridov, Yu.F.; Skulkin, N.N.; Ul'yanov, V.N.; Fedorov, Yu.D. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). 1977. 14p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

Some problems in developing an on-line system for automating technological measurements and experimental data acquisition at the BOR-60 reactor are discussed. Main features of the information flow at the reactor are considered. A computerized subsystem is described which consists of the KIPARIS-2 measuring and transmitting apparatus coupled with the BESM-3M or BESM-4M multi-user computer. Major characteristics of the subsystem are given. The subsystem has been in operation at the BOR-60 reactor for more than a year and proved a high degree of efficiency and reliability.

- 2348 (NIIAR-P--240) EFFICIENCY OF DIFFERENT TECHNIQUES OF PHYSICAL FLATTENING BY FUEL WHILE SELECTION OF OPTIMUM ARRANGEMENT OF LARGE FAST REACTOR CORE. Grachev, E.A.; Deinega, N.L.; Mitin, A.M. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). 1974. 19p. (In Russian). Dep. NTIS (US Sales Only), PC A02/MF A01.

Results are given of calculations for selecting the parameters of the large fast breeder reactor core (1500 Mw) operating on oxide fuel with a sodium coolant. A complex optimum criterion was selected for energy

intensity, energy distribution, breeding ratio and critical load factor, run duration, burning, reactivity variations, influence of CV3, fuel overloads, and calculated fuel expenses. The effectivities of various methods for physical grading of fuel (enrichment and composition) were examined in accordance with the optimum criterion. Parameters of reactor cores optimum arrangements are presented. Continuous reactor operation during 0.8-1.0 yr. at energy intensity more than 400 kW was shown to be essential for attaining the optimum chosen.

- 2349 (NIIR-P--255) PROGRAM FOR CALCULATION OF A NONSTATIONARY TEMPERATURE FIELD FOR A CYLINDRICAL FUEL ELEMENT. Antipin, G.K. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). 1975. 21p. (In Russian). Dep. NTIS (US Sales Only), PC A01/MF A01.

A programme is described for calculation of nonstationary field of a trilaminar cylindrical fuel element sprinkled by a non-compressible coolant. The field is described by the three differential energy equations, namely for fuel, shell and coolant. The external layer of the fuel element is a shell, the intermediate-contact sublayer and internal layer are fuel. A system of differential equations in partial derivatives is solved through a method of finite differences with assumption adopted for fast reactors. The algorithm is represented by ALGOL-60 programme serving for calculation of the nonstationary temperature field through a pre-set time pitch. The programme and its block diagram are presented. The programme is well-adjusted and under tentative operation. It is used, in particular, for calculation of nonstationary processes in the core of fast reactors with a sodium coolant.

- 2350 (NIEFA-A--0340) ELECTROMAGNETIC PUMPS FOR FAST REACTORS PRIMARY COOLING CIRCUITS. Andreev, A.M.; Bezgachev, E.A.; Karasev, B.G.; Kirillov, I.R.; Ogorodnikov, A.P.; Semikov, G.T. (Nauchno-Issledovatel'skij Inst. Elektrofizicheskoy Apparatury, Leningrad (USSR)). 1977. 24p. (In Russian). Dep. NTIS (US Sales Only), PC A03/MF A01.

The review of the electromagnetic pump development for the main loops of the powerful atomic energy stations with fast neutron reactors is presented. The calculation results of the cylindrical pump at the pressure of 9×10^5 Pa and Na consumption of $15500 \text{ m}^3/\text{h}$ are given. Its design is described, the comparison of three pump versions, differing by the coil cooling method: water, Na-K, nitrogen is considered. The design and insulating material choosing, cavitation aspects and pump electric power at emergency heating conditions are discussed.

- 2351 (NIEFA-A--0345) EXPERIMENTAL STUDY RESULTS OF THE TSLIN-5/700 CYLINDRICAL LINEAR INDUCTION PUMP. Andreev, A.M.; Karasev, B.G.; Kirillov, I.R.; Ogorodnikov, A.P.; Ostapenko, V.P.; Semikov, G.T. (Nauchno-Issledovatel'skij Inst. Elektrofizicheskoy Apparatury, Leningrad (USSR)). 1977. 22p. (In Russian). Dep. NTIS (US Sales Only), PC A03/MF A01.

The results of an experimental investigation of a cylindrical induction pump TSLIN-5/700 with the following operating characteristics are presented: pressure 5×10^5 Pa, current intensity 770 A, flow rate $700 \text{ m}^3/\text{hr}$, power output 209 kW, efficiency 45%. The investigation has been carried out on a liquid-metal stand at a sodium temperature of 350 deg

C. The operating and start-up characteristics of the pump different voltages are obtained. The thermal regime of the winding and the pump operation on parallel branches are investigated. The operability of the pump design, in which the thermal contact between the channel wall and the magnetic guide is ensured by the difference in the coefficients of linear expansion of the channel and the magnetic guide with the clamping rings have been confirmed. The pump operates steadily without any signs of cavitations at a low gauge pressure (up to 0.05×10^5 Pa) at a channel speed of 13 m/s. On the whole, the investigations confirm the possibility of constructing high-productivity and high-efficiency electromagnetic pumps capable of satisfying the operating requirements in the main loops of nuclear reactors.

- 2352 (NOR--3686) REVIEW OF THE UK FAS REACTOR PROGRAM: MARCH 1978. Smith, R.D. (UKAEA Risley Nuclear Power Development Establishment). Mar 1978. 47p. Dep. NTIS (US Sales Only), PC A03/MF A01.

The UK fast reactor program is briefly reviewed under the following section headings: (1) engineering development, (2) materials development, (3) chemical engineering/sodium technology, (4) fast reactor fuel, (5) fuel cycle, (6) safety, and (7) performance studies.

- 2353 (NP--21709) SOME TOPICS OF SODIUM TECHNOLOGY RESEARCH AND DEVELOPMENT IN P.N.C., JAPAN. Ishibashi, E. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). [nd]. 26p. Dep. NTIS (US Sales Only), PC A03/MF A01.

LMFBR TYPE REACTORS; REACTOR TECHNOLOGY; SODIUM; JOYO REACTOR; MONJU REACTOR; RESEARCH PROGRAMS

- 2354 (NP--22742) PRESSURE PULSE PROPAGATION IN PIPING SYSTEMS. FINAL REPORT. Cagliostro, D.J.; Wiersma, S.J.; Florence, A.L. (Stanford Research Inst., Menlo Park, CA (USA)). Jun 1975. 123p. Stanford Research Inst., Menlo Park, CA.

Experiments were conducted in which well-characterized pulses were generated in water-filled stainless steel piping systems consisting of a straight section of pipe, an open rectangular loop, and a closed rectangular loop. In some of the experiments, the closed loop was fitted with a standoff pipe that was filled with either water or air. The pulse shapes were typical of those expected in the secondary piping system as a result of a sodium-water reaction in a Liquid Metal Fast Breeder Reactor. The pipe thickness and diameter and the elbow radius are those typical of the Clinch River Fast Breeder Reactor reduced to 1/8-scale. Elastic piping response was stimulated (no plastic deformations). Pulse propagation behavior was monitored by pressure transducers installed in the piping wall at key locations. The overall results show that pulse propagation behavior exhibits (1) reduction of peak pressure and impulse after traversing an elbow, (2) reduction of peak pressure, but not of impulse, after passing a filled standoff pipe, (3) annihilation when meeting an empty standoff pipe, and (4) simple pressure addition when meeting a similar pulse.

- 2355 (NRPB-R--73) INFLUENCE OF THE PHYSICO-CHEMICAL FORM OF THE AEROSOL ON THE RADIOLOGICAL CONSEQUENCES OF NOTIONAL ACCIDENTAL RELEASES OF RADIOACTIVITY FROM A

FAST BREEDER REACTOR. Kelly, G.N.; Jones, J.A.; Simmonds, J.R. (National Radiological Protection Board, Harwell (UK)). Jan 1979. 147p. Dep. NTIS (US Sales Only), PC A08/MF A01.

The radiological consequences of a wide range of notional accidental releases from a 1300 MW(e) LMFBR (Liquid Metal-cooled Fast Breeder Reactor) were assessed in a study published by the National Radiological Protection Board (NRPB) in 1977. In that study representative values were in general adopted for each of the important parameters while recognising that in reality they could vary considerably. The present study is concerned with the sensitivity of the predicted consequences to the physico-chemical form of the released aerosol. Of particular interest is the importance of a mixed sodium-transuranium element aerosol which may be formed in accidental releases of activity from sodium cooled FBRs. Two significant findings emerge from the study. First the predicted consequences in general are relatively insensitive to the range of physico-chemical forms analysed. For generic assessments therefore it is sufficient to assume the properties of the aerosol adopted in the initial study (1 μ m AMAD and each element in the oxide form); the exception concerns the estimation of the incidence of early morbidity, and to a lesser extent early mortality, but only for a limited range of release composition. The second finding is that the radiological consequences are not, contrary to what might have been expected, significantly increased for the release of a mixed sodium-element aerosol.

2356 (NTIS/PS--79/0208) **CORROSION OF STAINLESS STEEL. VOLUME 2. 1977-MARCH, 1979 (A BIBLIOGRAPHY WITH ABSTRACTS). REPORT FOR 1977 MARCH 1979.** Reed, W.E. (National Technical Information Service, Springfield, VA (USA)). Mar 1979. 135p. NTIS PC N01/MF N01.

Federally-funded research on stainless steel composition, mechanical properties, structure, and testing in relation to corrosion is cited. The majority of research concerns nuclear reactor materials. In addition, corrosion studies in coal gasification plants, rocket propellant tanks, underwater cables, batteries, and geothermal systems are included. (This updated bibliography contains 129 abstracts, 55 of which are new entries to the previous edition.)

2357 (NUREG--0262) **STUDY OF DRYOUT HEAT FLUXES IN BEDS OF INDUCTIVELY HEATED PARTICLES.** Dhir, V.K.; Catton, I. (California Univ., Los Angeles (USA). School of Engineering and Applied Science). Feb 1977. 92p. NTIS \$5.00.

Experimental observations of the dryout heat fluxes for inductively heated particulate beds have been made. The data were obtained when steel and lead particles in the size distribution 295-787 microns were placed in a 4.7 cm diameter pyrex glass jar and inductively heated by passing radio frequency current through a 13.3 cm diameter multi-turn work coil encircling the jar. Distilled water, methanol and acetone were used as coolants in the experiments, while the bed height was varied from 1.0 to 8.9 cm. Different mechanisms for the dryout in deep and shallow beds have been identified. Dryout in shallow beds is believed to occur when the vapor velocity in the gas jets exceeds a certain critical velocity at which choking of the vapor occurs, leading to obstruction in the flow of the liquid toward the bed. However, deep beds dry out when

gravitational force can no longer maintain a downward coolant flow rate necessary to dissipate the heat generated in the bed. The heat flux data of the investigation and that from two previous investigations made at Argonne Laboratory and at UCLA have been correlated with semi-theoretical correlations based on the proposed hydrodynamic models. The deep and shallow bed correlations are used to predict the bed height at which transition from deep to shallow bed would occur. An application of the study has been made to determine the maximum coolable depths of the core debris as a function of the particle size, bed porosity and decay heat.

2358 (NUREG--0286) **STUDY OF THE REACTIVITY EFFECT OF BUBBLE COLLAPSE IN MOLTEN LIQUID METAL COOLED REACTORS.** Munno, F.J.; Sheaks, D.J. (Maryland Univ., College Park (USA). Dept. of Chemical and Nuclear Engineering). Aug 1977. 60p. NTIS \$4.50.

The reactivity effect of void collapse on a Core Disruptive Accident in a sodium free fast reactor core was studied. A two-dimensional transport theory computer code was employed to obtain a reactivity model which incorporated random distributions of voids in spatial location and bubble size. This model was implemented into VENUS-II and several excursions were investigated for the purpose of assessing the effect of bubble collapse reactivity on excursion energy. In addition, comparisons were made between the VENUS-II results obtained using the transport reactivity model and those obtained using a reactivity model developed from the Behrens' formulation of the leakage effects due to the presence of bubbles. The results indicate that excursion energy is sensitive to the shape of the reactivity versus void fraction function and that reactivity models based strictly on leakage effects are probably conservative compared to models which include both self-multiplication and leakage.

2359 (NUREG--0330(Vol.3)(No.1)) **STATUS SUMMARY REPORT ADVANCED REACTOR SAFETY RESEARCH. BUFF BOOK 2. (Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Regulatory Research).** 27 Feb 1979. 85p. Nuclear Regulatory Commission, Washington, DC.

Information for monitoring and controlling the progress of safety research projects associated with the breeder reactor program is presented. Data is utilized pertaining to project schedules, cost, and status which have been integrated into a network-based management information system. As part of this system's initial development, information has been collected and analyzed from the Argonne National Laboratory, Brookhaven National Laboratory, Battelle Columbus Laboratories, Sandia Laboratories, several universities, and the RES staff. Cost information has been coordinated with the Controller's Office.

2360 (NUREG--0344) **CONDENSATION OF FUEL ONTO THE ABOVE-CORE STRUCTURE DURING AN LMFBR CORE-DISRUPTIVE ACCIDENT.** Erdman, C.A.; Reynolds, A.B. (Virginia Univ., Charlottesville (USA)). Aug 1977. 153p. NTIS \$8.00.

Condensation of a pure, saturated vapor onto a vertical, melting substrate is analyzed for both one- and two-material situations. Examination of the one-material situation indicates that the solution to the full transient condensation-induced melting problem may be approximated by using a transient,

conduction-only model for short times and a steady-state, flowing-film model for long times. This concept is extrapolated to the two-material situation in order to obtain a simulation of the transient solution. The models are applied to the specific case of uranium dioxide condensing onto solid stainless steel. Condensate solidification occurs for this pair of materials; however, this solidification may be neglected without introducing a serious error in the other phase-change rates. The condensation heat flux for this pair of materials is a very weak function of the initial substrate temperature and the vapor temperature. The results of this analysis have applications in the area of LMFBR accident analysis.

- 2361 (NUREG--0358) SAFETY EVALUATION REPORT. FAST FLUX TEST FACILITY. PROJECT NO. 448. (Nuclear Regulatory Commission; Washington, DC (USA). Office of Nuclear Reactor Regulation). 1 Aug 1978. 399p. NTIS \$13.25.

Information on the safety of the FFTF Reactor is presented under the following chapter headings: site characteristics; design of structures, components, equipment, and systems; reactor; reactor coolant system and connected systems; engineered safety features; electric power; auxiliary systems; radioactive waste management systems; radiation protection; conduct of operations; initial test programs; accident analysis; and quality assurance.

- 2362 (NUREG--0358(Suppl.1)) SAFETY EVALUATION REPORT RELATED TO OPERATION OF FAST FLUX TEST FACILITY. SUPPLEMENT NO. 1. (Nuclear Regulatory Commission, Washington, DC (USA). Office of Nuclear Reactor Regulation). May 1979. 160p. NTIS, PC A08/MF A01.

This supplement provides (1) the staff's evaluation of additional information received since issuance of the Safety Evaluation Report regarding previously identified uncompleted review items, (2) a discussion of comments made by the ACRS in its report of November 8, 1978, and (3) the staff's evaluation of additional or revised information related to new or old issues that have arisen since the issuance of the Safety Evaluation Report.

- 2363 (NUREG--0387) CORROSION CRITIQUE OF THE 2 1/4 CR-1 MO STEEL FOR LMFBR STEAM GENERATION SYSTEM APPLICATIONS. Zima, G.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jul 1977. Contract EY-76-C-06-1830. 79p. NTIS, MF A01.

Portions of document are illegible.

The unstabilized ferritic steel of nominal composition, 2 1/4Cr-1Mo, has been proposed for critical structural assignments in LMFBR powerplants, specifically: the tubing, tubesheet and shell of the evaporator and superheater components. The interest in this steel has been based on a presumably favorable general corrosion property spectrum, acceptable mechanical properties and fabricability, and certain economies associated with the low alloy content. This report is an attempt at a general corrosion assessment for the 2 1/4Cr-1Mo steel and an identification of corrosion problem areas potential to this steel from the sodium and water/steam systems of the proposed working environment. There is a considerable area of uncertainty in the sodium-side response of 2 1/4Cr-1Mo steel, centered in the loss and redistribution of carbon during long-term exposure to sodium of various impurity backgrounds. It is submitted that present evidence relating to the water/steam-side

corrosion behavior of the 2 1/4Cr-1Mo steel, under nominal and conceivable perturbed environmental conditions, constitutes the principal concern for the proposed LMFBR powerplant applications of this steel. It is suggested that this unfavorable corrosion aspect represents an inherent limitation of the low alloy content of this steel, probably largely independent of melting and processing recourses, and it is a sufficient basis to question the incentive for a continuation of the collateral studies of this steel for the proposed LMFBR steam generation system assignments.

- 2364 (NUREG--0391) PRIMARY AEROSOL PARTICLE-SIZE DISTRIBUTION FROM HOMOGENEOUS NUCLEATION CONDENSATION AND PARTICLE GROWTH. Kennedy, M.F.; Erdman, C.A.; Reynolds, A.B.; Walter, A.E. (Virginia Univ., Charlottesville (USA). Dept. of Nuclear Engineering and Engineering Physics). Nov 1977. 95p. (UVA--529106/NEEP-77/102). NTIS \$6.00.

The process of forming aerosols by homogeneous nucleation condensation followed by condensation growth was studied as a potential source of submicrometre aerosols in LMFBR accidents. A model was developed for calculating particle size distributions following nucleation and growth. This model was applied to experiments at ORNL where UD₂ fuel pellets were partially vaporized in an argon atmosphere and the resulting primary particle size distribution was measured. The low range of particle sizes observed in the ORNL tests could be reproduced by the model by assuming large mixing rates between the fuel vapor and the argon, thus indicating that homogeneous nucleation condensation was likely the source of the small particles observed.

- 2365 (NUREG--0457(Rev.0)) QUALIFICATION TESTING PROGRAM PLAN FOR SIMMER. A COMPUTER CODE FOR LMFBR DISRUPTED CORE ANALYSIS. Basdekas, D.L.; Silberberg, M.; Curtis, R.T.; Kelber, C.N. (Nuclear Regulatory Commission, Washington, DC (USA). Div. of Reactor Safety Research). Jul 1978. 48p. NTIS \$5.25.

The objective of SIMMER qualification testing program is to assure that the mathematical models and input parameters are derived from experimental data, which, on the basis of criteria still to be established, are representative of the phenomena and processes governing the progression of a CDA in an LMFBR. At the present time, the work to meet this objective can be classified into three general task areas as they relate to the use of SIMMER in CDA analysis: (1) The whole-core energetic disassembly accident, or the 'vessel problem': The objective here is to predict the partition of the total energy release, by a postulated severe power excursion, between the primary containment and the energy absorbed through nondestructive dissipative processes. (2) Single subassembly accident: The objective here is to determine the pertinent phenomena and to develop the capability to assess the significance and likelihood that such accidents might propagate to involvement of larger fraction of the core. (3) The whole-core transition phase accident: The objective here is to advance the understanding of the phenomena and processes involved, so that reliable predictions can be made of the possible consequences of a CDA and the potential for further nuclear excursions through recriticality.

- 2366 (NUREG--0657) REVIEW AND EVALUATION OF THE NUCLEAR REGULATORY COMMISSION SAFETY

RESEARCH PROGRAM FOR FISCAL YEAR 1981: REPORT TO THE CONGRESS OF THE UNITED STATES OF AMERICA. (Nuclear Regulatory Commission, Washington, DC (USA). Advisory Committee on Reactor Safeguards). Feb 1980. 53p. NTIS.

Information is presented concerning implications of the Three Mile Island accident and new directions in research: systems engineering; LOFT; code development; fuel behavior; primary system integrity; seismic, engineering, and site safety; advanced reactors; reactor environmental effects; fuel cycle; waste management; safeguards; risk assessment; and improved reactor safety.

- 2367 (NUREG/CP--0004(Vol.1), pp 85-99) ZERO-POWER BASIS OF FAST REACTOR DOSIMETRY. Sanders, J.E. (Atomic Energy Establishment, Winfrith, Eng.). 1977.

From 2. ASTM-EURATOM symposium on reactor dosimetry: dosimetry methods for fuels, cladding and structural materials; Palo Alto, CA, USA (2 Oct 1977).

Predictions of reaction rates, atomic displacements, and gamma-ray energy deposition in the Prototype Fast Reactor are based on cross-section data and calculation methods validated against the results of zero-power experiments. The paper reviews work in Zebra relevant to this dosimetry, including neutron spectrometry, power mapping, foil activations within core heterogeneities, and measurements with thermoluminescent detectors. Comparisons of experiment and calculation are discussed in relation to the accuracies required to meet materials testing objectives.

- 2368 (NUREG/CP--0004(Vol.3), pp 1141-1175) STATUS REPORT ON DOSIMETRY BENCHMARK NEUTRON FIELD DEVELOPMENT, CHARACTERIZATION, AND APPLICATION. Fabry, A. (SCK/CEN, Mol, Belgium); Grundl, J.A.; McElroy, W.N.; Lippincott, E.P.; Farrar, H. IV. 1977.

From 2. ASTM-EURATOM symposium on reactor dosimetry: dosimetry methods for fuels, cladding and structural materials; Palo Alto, CA, USA (2 Oct 1977).

The report attempts to present a brief, but comprehensive review of the status and future directions of benchmark neutron field development, characterization and application in perspective with two major objectives of reactor dosimetry: (1) fuel fission rate and burn-up passive monitoring, and (2) correlation of materials irradiation damage effects and projection to commercial power plants. The report focuses on the Light Water Reactor and Fast Breeder Reactor program needs.

- 2369 (NUREG/CP--0004(Vol.3), pp 1267-1278) FAST REACTOR FISSION YIELD MEASUREMENT PROGRAM AT THE IDAHO NATIONAL ENGINEERING LABORATORY. Maeck, W.J.; Emel, W.A.; Delmore, J.E. (Allied Chemical Corp., Idaho Falls, ID). 1977.

From 2. ASTM-EURATOM symposium on reactor dosimetry: dosimetry methods for fuels, cladding and structural materials; Palo Alto, CA, USA (2 Oct 1977).

The INEL fission yield measurement program and the current status of the results are reviewed. Fast reactor fission yields for ^{233}U , ^{235}U , ^{238}U , ^{239}Pu , ^{241}Pu , and ^{237}Np obtained for samples irradiated in Row-8 of EBR-II have been reported. The neutron spectrum associated with the reported yields has been reported. Measurement of fast reactor fission yields for ^{240}Pu , ^{242}Pu , and ^{241}Am is in progress. To supplement these data, and to provide additional correlation points relative to the effect of neutron energy on fission yields, a

series of samples previously irradiated in the axial blanket and Row-2 to 4 of EBR-II has been obtained for analysis. Future efforts relative to recommending and proof-testing methods for the analysis of burnup for FBR fuels are discussed.

- 2370 (NUREG/CP--0004(Vol.3), pp 1279-1287) STUDIES RELATIVE TO THE CORRELATION OF FISSION YIELDS WITH NEUTRON ENERGY. Maeck, W.J. (Allied Chemical Corp., Idaho Falls, ID). 1977.

From 2. ASTM-EURATOM symposium on reactor dosimetry: dosimetry methods for fuels, cladding and structural materials; Palo Alto, CA, USA (2 Oct 1977).

The status of fission yield data is briefly reviewed. Problems associated with the evaluation and compilation of fast reactor fission yields are discussed. A new approach for evaluating the change in fission yields with neutron energy is presented. Based on a comparison of relative isotopic abundances rather than absolute yield data, changes of only a few percent in the isotopic abundances of the neodymium isotopes as a function of neutron energy are easily detected.

- 2371 (NUREG/CP--0004(Vol.3), pp 1289-1306) REFERENCE AND STANDARD BENCHMARK FIELD CONSENSUS FISSION YIELDS FOR U.S. REACTOR DOSIMETRY PROGRAMS. Gilliam, D.M. (National Bureau of Standards, Washington, DC); Helmer, R.G.; Greenwood, R.C.; Rogers, J.W.; Heinrich, R.R.; Popek, R.J.; Kellogg, L.S.; Lippincott, E.P.; Hansen, G.E.; Zimmer, W.H. 1977.

From 2. ASTM-EURATOM symposium on reactor dosimetry: dosimetry methods for fuels, cladding and structural materials; Palo Alto, CA, USA (2 Oct 1977).

Measured fission product yields are reported for three benchmark neutron fields--the BIG-10 fast critical assembly at Los Alamos, the CFRMF fast neutron cavity at INEL, and the thermal column of the NBS Research Reactor. These measurements were carried out by participants in the Interlaboratory LMFBR Reaction Rates (ILRR) program. Fission product generation rates were determined by post-irradiation analysis of gamma-ray emission from fission activation foils. The gamma counting was performed by Ge(Li) spectrometry at INEL, ANL, and HEDL; the sample sent to INEL was also analyzed by NaI(Tl) spectrometry for Ba-140 content. The fission rates were determined by means of the NBS Double Fission Ionization Chamber using thin deposits of each of the fissionable isotopes. Four fissionable isotopes were included in the fast neutron field measurements; these were U-235, U-238, Pu-239, and Np-237. Only U-235 was included in the thermal neutron yield measurements. For the fast neutron fields, consensus yields were determined for three fission product isotopes--Zr-95, Ru-103, and Ba-140. For these fission product isotopes, a separately activated foil was analyzed by each of the three gamma counting laboratories. The experimental standard deviation of the three independent results was typically $\pm 1.5\%$. For the thermal neutron field, a consensus value for the Cs-137 yield was also obtained. Subsidiary fission yields are also reported for other isotopes which were studied less intensively (usually by only one of the participating laboratories). Comparisons with EBR-II fast reactor yields from destructive analysis and with ENDF/B recommended values are given.

- 2372 (NUREG/CP--0004(Vol.3), pp 1307-1321) EDB-II VALIDATED, KEY FISSION PRODUCT YIELDS

FOR FAST REACTOR APPLICATION. Kellogg, L.S.; Davis, R.I.; Lippincott, E.P.; Ruggles, J.M. (Hanford Engineering Development Lab., Richland, WA). 1977.

From 2. ASTM-EURATOM symposium on reactor dosimetry: dosimetry methods for fuels, cladding and structural materials; Palo Alto, CA, USA (2 Oct 1977).

Relative fission yields were measured for three different locations in the row 4 "Test Region" of the EBR-II reactor. Correlation of the relative fission yields to the measured average energy (anti E) and the measured ^{137}Cs $^{238}\text{U}/^{235}\text{U}$ spectral indices have been made. The measured relative fission yields for selected fission products from ^{235}U , ^{238}U , ^{239}Pu and ^{237}Np have been compared with those values reported by the Interlaboratory Reaction Rate (ILRR) program, EBR-II fast reactor yields from destructive analysis and summation, and the March 1977 version of ENDF/B-V.

2373 (NUREG/CP--0007, pp I.1.1-I.1.28) REVIEW OF HIGH TEMPERATURE MEASUREMENT TECHNIQUES FOR LMFBR SAFETY EXPERIMENTS. Starkovich, V.S. (Los Alamos Scientific Lab., NM). Oct 1979.

From Reactor Safety Instrumentation Review Group meeting; Silver Spring, MD, USA (24 Jul 1979).

A summary of the latest developments in ultrasonic thermometry and a discussion of the potential applications of pyrometry to LMFBR safety experiments is presented. A brief discussion of several high temperature measurement techniques which were presented at the 25th Instrumentation Society of America meeting in Anaheim, California, May 6 to 9, is also included.

2374 (NUREG/CR--0001) ROD-BUNDLE THERMAL-HYDRAULIC ANALYSIS USING BOUNDARY-FITTED COORDINATE SYSTEM. Sha, W.T.; Thompson, J.F. (Argonne National Lab., IL (USA)). Jan 1979. Contract W-31-109-ENG-38. 45p. (ANL--78-1). Dep. NTIS, PC A03/MF A00.

A formulation under development for a three-dimensional, time-dependent, compressible single-phase rod-bundle thermal-hydraulic analysis using boundary-fitted coordinate systems is discussed. The governing equations of conservation of mass, momentum, and energy are treated as a boundary-value problem in space and an initial-value problem in time. The numerical solutions can be in either implicit or explicit form, each with a choice of central or upwind convective derivatives. Two general types of coordinate systems are considered, analogous to rectangular and cylindrical systems. Complicated cross-sectional rod-bundle geometries, such as hexagonal fuel assemblies used in the LMFBR, can be transformed into either rectangular or cylindrical coordinates with uniform mesh.

2375 (NUREG/CR--0002) BUBBLE BEHAVIOR IN LMFBR CORE DISRUPTIVE ACCIDENTS. ANNUAL REPORT, JULY 1, 1976--SEPTEMBER 30, 1977. Erdman, C.A.; Reynolds, A.B. (Virginia Univ., Charlottesville (USA). Dept. of Nuclear Engineering and Engineering Sciences). Dec 1977. 126p. (UVA--529106/NE77/103). NTIS \$7.25.

Analytical research in three major areas is described: (1) particle production by homogeneous nucleation, (2) fuel condensation on above-core structure, and (3) small particle production by hydrodynamic fragmentation. Preliminary results on minimum expected particle sizes for one stage of LMFBR core disruptive accidents are presented along with

work on time scales for fragmentation and a status report of current efforts.

2376 (NUREG/CR--0011) LMFBR FUEL ANALYSIS. TASK A: OXIDE FUEL DYNAMICS. FINAL REPORT, OCTOBER 1, 1976--SEPTEMBER 30, 1977. Dhir, V.K.; Doshi, J.; Frank, M.; Hauss, B.; Kastenber, W.E.; Wong, K. (California Univ., Los Angeles (USA). Dept. of Chemical, Nuclear, and Thermal Engineering). Oct 1977. 132p. NTIS \$7.25.

The study presented deals with several areas of uncertainty in the analysis of the unprotected overpower transient for the Clinch River Breeder Reactor. These areas of uncertainty include the time, place, and mode of fuel pin failure; pre-failure fuel motion; fuel freezing, plugging, and plate-out following pin failure; and the potential for re-criticality. Internal molten fuel motion prior to pin failure was found to be sensitive to ramp rate and burnup. The strain-limit fuel failure criterion was found to be inappropriate for analysis based on existing data. The coupling of pre-transient- and transient-induced stresses tended to force the failure location towards the core midplane.

2377 (NUREG/CR--0012) LMFBR FUEL ANALYSIS. TASK B: POST-ACCIDENT HEAT REMOVAL. FINAL REPORT, OCTOBER 1, 1976--SEPTEMBER 30, 1977. Castle, J.; McDonough, J.M.; Ostrander, N.C.; Catton, I. (California Univ., Los Angeles (USA). School of Engineering and Applied Science). Oct 1977. 157p. NTIS \$8.00.

This task B annual report focuses on the interaction of concrete structures and core debris which may be released in a LMFBR core disruptive accident. A heat transfer calculational technique is described and applied to the exchange of energy from hot core debris to the concrete. Predictions are compared to several test conditions for which data are available. Calculations are made of the extent to which debris can penetrate the structure prior to solidification. The Clinch River plant and the FFTF are used as two examples in the calculations. The numerical characteristics of the Sandia INTER and Westinghouse WAYER post-accident heat removal codes are scrutinized. An evaluation is made of the heat transfer characteristics exhibited in material interaction tests conducted at the Aerospace Corporation.

2378 (NUREG/CR--0013) LMFBR FUEL ANALYSIS. TASK C: RELIABILITY ASPECTS OF LMFBRs. FINAL REPORT, OCTOBER 1, 1976--SEPTEMBER 30, 1977. Basti, W.; Kastenber, W.E. (California Univ., Los Angeles (USA). Dept. of Chemical, Nuclear, and Thermal Engineering). Oct 1977. 63p. NTIS \$5.25.

An analysis is presented for the availability of the electrical power supplies upon reactor shutdown. Successful power supply is defined in terms of the ability of the associated pumps (pump motors) to provide forced circulation and to deliver sufficient feedwater for proper cooldown of the core. Previous investigations of the reliability of the CRBR shutdown heat removal system concentrated on the mechanical systems and/or did not yet consider the diverse power supply. The shutdown heat removal system (SHRS) is discussed in the light of the availability of the electrical power systems, depending upon various types of initiating events. The unavailabilities of the essential power distribution and power supply buses are estimated, so that they can easily be used in connection with analyses of the entire SHRS.

Further estimates include mechanical failure of the pumps.

extensive subassembly voiding.

- 2379 (NUREG/CR--0015) ANALYSIS OF AEROSOL PARTICLES UNDERGOING GRAVITATIONAL AGGLOMERATION. ANNUAL PROGRESS REPORT, DECEMBER 15, 1976--SEPTEMBER 30, 1977. Loyalka, S.K. (Missouri Univ., Columbia (USA). Coll. of Engineering). Nov 1977. 66p. NTIS \$5.25.

Studies are reported of aerosol behavior following an LMFBR reactor core disruption. Highlights of progress in theoretical area have consisted of a clear and consistent formulation of the dynamical equations of motion for the colliding particles. Relevant Drag forces have been identified and methods for obtaining such forces are discussed. A computer program that determines the collision efficiency by solving the equations of motion and numerical experimentation is operational. At present, the Stokesian Drag forces with slip corrections are being used, but other drag forces will be also incorporated. Results obtained to date indicate that for the range of particle sizes studied so far the collision efficiency decreases with decrease in density. An extensive review of previous experimental work on collision efficiency in the atmospheric sciences was completed. Use of both the macroscopic and the microscopic type of experiments is recommended. The former include laser scattering type of work and the latter include similitude type of experiments. The similitude type of experiments is especially suitable for studying the density and shape effects on the collision efficiency, and the necessary apparatus has been assembled at the UMC. Preliminary experimental parameters have been identified, and initial results are expected to be available in the near future.

- 2380 (NUREG/CR--0016) PARAMETER SENSITIVITY STUDY IN THE DYNAMIC SIMULATION OF A SODIUM COOLED BREEDER REACTOR POWER PLANT. Kolb, G.J. (Arizona Univ., Tucson (USA). Coll. of Engineering). Jan 1978. 71p. NTIS \$5.25. Thesis.

The time domain sensitivity measure, peak sensitivity, was employed in studying the dynamics of a Clinch River Breeder Reactor Power Plant (CRBRP) digital simulator. With its use the most "safety sensitive" parameters in the plant model were identified for particular anticipated transients. Examples of such parameters include the Doppler feedback coefficient, fraction of power generated in the core, and the two phase flow friction multiplier. A method of putting first order bounds on a particular power plant transient due to parameter estimate error is also presented. This information is valuable since the CRBRP simulator was constructed with the intent of studying the safety of the system.

- 2381 (NUREG/CR--0018) POST-FAILURE PHENOMENA IN LMFBR TOP ACCIDENTS. FINAL REPORT, OCTOBER 1, 1976--SEPTEMBER 30, 1977. Erdman, C.A.; Bradley, D.R.; Kelley, J.L. (Virginia Univ., Charlottesville (USA). School of Engineering and Applied Science). Nov 1977. 75p. (UVA--529058/NEEP77/103). NTIS \$6.00.

Work in the area of post-failure phenomena in LMFBR TOP accidents is reported. Several different TOP-related topics are treated, including radial dependence of transient fission gas behavior in LMFBR pins; modeling of flowing-sodium pin failure tests in TREAT; gas plenum region pressurization due to sodium vaporization; and the effect of ejection rates of molten fuel from pins on the potential for

- 2382 (NUREG/CR--0076) EVALUATION OF MATERIALS FOR CRBRP CORE RETENTION. ANNUAL PROGRESS REPORT. Swanson, D.G. (Aerospace Corp., Los Angeles, CA (USA). Materials Science Lab.). Oct 1977. 152p. (ATR--77(7608)=2). NTIS \$8.00.

A series of experiments were performed to study the interactions expected in a hypothetical core meltdown accident between molten core debris, liquid sodium, and reactor materials. In "transient" molten core debris experiments, either UO_2 or stainless steel was melted in an electric arc furnace, poured on a specimen to simulate thermal shock, and allowed to cool. Later, in combined transient-sustained heating experiments, the UO_2 or steel layer was remelted in another arc furnace and maintained in a molten state in order to study the penetration of molten core debris into the specimen. During the liquid sodium experiments, sodium was poured on the material being tested and was then maintained at a predetermined temperature in a single operation. In both the molten core debris and liquid sodium experiments, interactions with MgO , basalt concrete, magnetic concrete, and two kinds of limestone concrete were investigated. Temperature data were obtained, and gas samples were analyzed in both series of experiments. In addition, penetration rates were determined in the molten core debris experiments. Some microstructural analysis has been completed but more remains to be done.

- 2383 (NUREG/CR--0084) AEROSOL MEASUREMENTS AND MODELING FOR FAST REACTOR SAFETY. QUARTERLY PROGRESS REPORT, OCTOBER 1--DECEMBER 31, 1977. Gieseke, J.A.; Jordan, H.; Lee, K.W.; Schmidt, E.W.; Vaishnavi, B.; Reed, L.D. (Battelle Columbus Labs., OH (USA)). May 1978. 44p. (BMT--1999). NTIS \$4.50.

Technical progress during the quarter was made in areas of HAARM-3 computer code improvements, comparisons of the code with the experimental data, BAM code evaluation, reference code development, and agglomerate characteristics measurements including calibration of the spiral duct centrifuge. Details of efforts directed toward the major technical areas are presented.

- 2384 (NUREG/CR--0115) EFFECTS OF ENERGY LOSSES OF ARBITRARY MAGNITUDE ON FUEL QUALITY, VAPOR DENSITY, AND BUBBLE DIAMETER IN AN IDEALIZED EXPANSION OF MIXTURES OF UO_2 AND NA. Bertini, H.W.; White, J.S. (Oak Ridge National Lab., TN (USA)). Jul 1978. Contract W-7405-ENG-26. 111p. (ORNL/NUREG/TM--185). Dep. NTIS, PC A06/MF A01.

A method is developed for calculating the characteristics of the expected bubble in capacitor-discharge UO_2 vaporization experiments in sodium. These characteristics are the quality of the vaporized UO_2 and its vapor density and the bubble diameter. Results are presented in tabular form for a parametric study covering the range of initial and final states to be expected in these experiments and for a variety of energy losses and mixtures of UO_2 and Na. These data can also be used for sensitivity studies of the fuel quality and vapor density in a bubble that might be formed from a hypothetical core-disruptive accident in a liquid-metal fast breeder reactor. In order to facilitate the calculations, idealized expansions are assumed, that is, expansions that are free of intervening structure and in which thermodynamic equilibrium exists at all times.

- 2385 (NUREG/CR--0117) LMFBR AEROSOL RELEASE AND TRANSPORT PROGRAM QUARTERLY PROGRESS REPORT, JANUARY--MARCH 1978. Kress, T.S. (Oak Ridge National Lab., TN (USA)). 25 Aug 1978. Contract W-7405-ENG-26. 62p. (ORNL/NUREG/TM--213). Dep. NTIS, PC A04/MF A01.

Progress is summarized in studies designed to investigate radionuclide release and transport from LMFBRs for reactor events of severity up to and including hypothetical core-disruptive accidents (HCDAs). Experimental program topics discussed include: recent Capacitor Discharge Vaporization tests in the CRI-III facility at its new FAST site; FAST facility installation progress; performance tests for the consumable electrode aerosol generator in NSPP and the U_3O_8 aerosol behavioral results accompanying the performance tests; development of a back-up "flame-spray" U_3O_8 aerosol generator for high densities; and progress in making a trial HCDA source term analysis.

- 2386 (NUREG/CR--0120) EFFECT OF ENERGY DENSITY ON AEROSOL YIELD AND PRIMARY PARTICLE SIZES PRODUCED BY THE CAPACITOR DISCHARGE VAPORIZATION (CDV) OF UO_2 . Wright, A.L.; Bertini, H.W.; Kress, T.S.; Tobias, M.L.; Kelly, M.J.; Parsly, L.F.; White, J.S. (Oak Ridge National Lab., TN (USA)). 31 Aug 1978. Contract W-7405-ENG-26. 54p. (ORNL/NUREG/TM--163). Dep. NTIS, PC A04/MF A01.

The report documents experimental results obtained in the CRI-III/CDV portion of the ORNL Aerosol Release and Transport (ART) Program, sponsored by the Advanced Reactor Safety Research Division of the Nuclear Regulatory Commission. In the CRI-III/CDV program, the capacitor discharge vaporization (CDV) technique is used to bring simulant LMFBR fuel samples (UO_2) to energy states representative of those calculated for severe hypothetical core-disruptive accidents (HCDAs). The high-energy molten UO_2 undergoes a free expansion in the CRI-III vessel, and the vapor produced converts to an aerosol. The report discusses the results of tests where the major objective was to establish the UO_2 aerosol mass made airborne (yield) and the size distribution of the primary aerosol as a function of the energy content of the UO_2 before expansion.

- 2387 (NUREG/CR--0137) PROMPT BURST 2NERGETICS EXPERIMENTS: URANIUM CARBIDE SERIES. PRELIMINARY RESULTS. Reil, K.O.; Young, M.F.; Camp, W.J.; Plitz, H. (Sandia Labs., Albuquerque, NM (USA)). Jun 1978. Contract EY-76-C-04-0789. 26p. (SAND--78-0752). Dep. NTIS, PC A03/MF A01.

A brief overview is presented of the results of the SG series of prompt burst energetics (PBE) experiments in the Sandia Laboratories Annular Core Pulse Reactor (ACPR). This series (PBE-SG, 2, and 3) involved single uranium-carbide fuel pins which were rapidly pulse heated in a stagnant sodium channel to extreme energy states characteristic of prompt burst conditions. The purpose of the PBE program is to characterize, in an integral pin geometry, the phenomena resulting in pressurization and conversion of thermal to mechanical energy. To this end temperature and pressure histories are directly monitored and the work energy imparted to the system constraint is measured using a movable piston technique. Knowing the neutron energy deposition, one is able to calculate the overall energy conversion efficiency of the experiment, and (somewhat indirectly) apportion the pressure generated between fuel and coolant vapor. Although only very preliminary analysis

has been performed, in the case of all three tests it appears that measured pressures cannot be explained on the basis of fuel vapor pressure alone. Further, in the test with the greatest energy deposition (SG2), peak pressures obtained were far greater than those obtained in any previous PBE test with uranium oxide fuel. Nevertheless overall thermal to mechanical energy conversion efficiency remains low for the experiment as a whole (although local efficiencies could be quite high).

- 2388 (NUREG/CR--0148) PHYSICS OF REACTOR SAFETY. QUARTERLY REPORT, JANUARY--MARCH 1978. (Argonne National Lab., IL (USA)). Jun 1978. Contract W-31-109-ENG-38. 52p. (ANL--78-52). Dep. NTIS, PC A04/MF A01.

This report summarizes work done on reactor safety, Monte Carlo analysis of safety-related critical assembly experiments, and planning of Safety-Related (ZPR) experiments. Work on reactor core thermal-hydraulic code development is also included.

- 2389 (NUREG/CR--0166) SUSTAINED MOLTEN STEEL/CONCRETE INTERACTIONS TESTS: A PRELIMINARY REPORT ON THE FEASIBILITY OF EXPERIMENTAL TECHNIQUES. Powers, D.A. (Sandia Labs., Albuquerque, NM (USA)). Jun 1978. Contract EY-76-C-04-0789. 52p. (SAND--77-1423). Dep. NTIS, PC A04/MF A01.

An experimental demonstration of the feasibility of using induction heating to maintain 200 kilograms of stainless steel molten while in contact with concrete is described. Data are presented to show that operation of an induction coil embedded in the concrete at 120 kW was sufficient to sustain temperatures near 1500°C in the melt for 10 minutes. Metal losses by gas induced expulsion from the crucible and oxidation may have been responsible for the inability to achieve higher temperatures. Techniques for monitoring melt temperatures, the compositions and temperatures of evolved gases, and concrete temperatures during the test are described.

- 2390 (NUREG/CR--0167) SURVEY OF EXISTING MOMENTUM EXCHANGE MODELS IN TWO-PHASE FLOWS FOR USE IN THE SIMMER FAST REACTOR SAFETY ANALYSIS CODE. Eisenhower, S.W. (Sandia Labs., Albuquerque, NM (USA)). Mar 1978. Contract EY-76-C-04-0789. 43p. (SAND--78-0167). Dep. NTIS, PC A03/MF A01.

Models which describe momentum exchange in two-phase, vapor-liquid flows are surveyed. These models are related to the momentum exchange coefficient formulations used in the SIMMER code. The result is a set of model-dependent exchange coefficients for various flow regimes. Criteria for flow regime transitions and experimental needs in momentum exchange modeling are also discussed.

- 2391 (NUREG/CR--0224) MULTIPHASE TRANSIENTS WITH COOLANT AND CORE MATERIALS IN LMFBR CORE DISRUPTIVE ACCIDENT ENERGETICS EVALUATIONS. Theofanous, T.G. (Purdue Univ., Lafayette, IN (USA)). School of Nuclear Engineering). Jun 1978. 298p. NTIS \$11.75.

An overview of core disruptive accidents in LMFBRs and an assessment of their potential for energetic consequences are presented. Areas in which current state-of-the-art is found inadequate are identified. In particular the early post disruptive period (called herein Transition to the Transition Phase) is identified as a key area having received inadequate attention to date. Further the idea of mild dispersal from a fully developed

Transition Phase at decay heat power levels does not appear justified. However considerable question exists as to the attainability of such full core pool under accident conditions. Specific recommendations for research work in relation to licensing needs are presented.

- 2392 (NUREG/CR--0240) SINGLE-PHASE PUMP MODEL FOR ANALYSIS OF LMFBR HEAT TRANSPORT SYSTEMS. Madai, I.K.; Cazzoli, E. (Brookhaven National Lab., Upton, NY (USA)). May 1978. Contract EY-76-C-02-0016. 52p. (BNL-NUREG--50859). Dep. NTIS, PC A04/MF A01.

A single-phase pump model for transient and steady-state analysis of LMFBR heat transport systems is presented. Fundamental equations of the model are angular momentum balance to determine transient impeller speed and mass balance (including thermal expansion effects) to determine the level of sodium in the pump tank. Pump characteristics are modeled by homologous head and torque relations. All regions of pump operation are represented with reverse rotation allowed. The model also includes option for enthalpy rise calculations and pony motor operation. During steady state, the pump operating speed is determined by matching required head with total load in the circuit. Calculated transient results are presented for pump coastdown and double-ended pipe break accidents. The report examines the influence of frictional torque and specific speed on predicted response for the pump coastdown to natural circulation transient. The results for a double-ended pipe break accident indicate the necessity of including all regions of operation for pump characteristics.

- 2393 (NUREG/CR--0244) BRENDA: A DYNAMIC SIMULATOR FOR A SODIUM-COOLED FAST REACTOR POWER PLANT. Hetrick, D.L.; Sowers, G.W. (Arizona Univ., Tucson (USA). Dept. of Nuclear Engineering). Jun 1978. 87p. (EES--010-3). NTIS \$6.00.

This report is a users' manual for one version of BRENDA (Breeder Reactor Nuclear Dynamic Analysis), which is a digital program for simulating the dynamic behavior of a sodium-cooled fast reactor power plant. This version, which contains 57 differential equations, represents a simplified model of the Clinch River Breeder Reactor Project (CRBRP). BRENDA is an input deck for DARE P (Differential Analyzer Replacement, Portable), which is a continuous-system simulation language developed at the University of Arizona. This report contains brief descriptions of DARE P and BRENDA, instructions for using BRENDA in conjunction with DARE P, and some sample output. A list of variable names and a listing for BRENDA are included as appendices.

- 2394 (NUREG/CR--0245) VAPOR EXPLOSION PHENOMENA WITH RESPECT TO NUCLEAR REACTOR SAFETY ASSESSMENT. Cronenberg, A.W.; Benz, R. (Idaho National Engineering Lab., Idaho Falls (USA)). Jul 1978. Contract EY-76-C-07-1570. 75p. (TREE--1242). Dep. NTIS, PC A04/MF A01.

Stringent licensing procedures for commercial nuclear reactor operation require an in-depth analysis of the phenomena associated with postulated reactor core overheating accidents. One aspect of nuclear reactor safety assessment is a prediction of the consequences of interaction between molten fuel and coolant, in which rapid heat transfer from the fuel may lead to explosive vaporization of the coolant. Some of the most recent theories relating to vapor explosion research for nuclear reactor safety assessment are critically reviewed and

assessed. Specifically, the spontaneous nucleation and pressure detonation models are discussed at length. In addition, modeling of the fragmentation of hot molten fuel upon contact with liquid coolant, and energy considerations for rapid fine-scale intermixing of such fragmented fuel with coolant are discussed in detail. 141 references.

- 2395 (NUREG/CR--0263) DEBRIS BED STUDIES AND EXPERIMENTS AT SANDIA LABORATORIES. Rivard, J.B. (Sandia Labs., Albuquerque, NM (USA)). Jul 1978. Contract EY-76-C-04-0789. 57p. (SAND--78-0299). Dep. NTIS, PC A04/MF A01.

An initial experiment simulating a 58 mm thick LMFBR fuel debris bed has been performed in liquid sodium in which the UO_2 particles composing the bed were internally heated by nuclear fission. The experiment was performed in the Annular Core Pulse Reactor (ACPR), as part of Sandia Laboratories' program examining the safety of advanced reactors. Supporting research included experiments utilizing other power sources and simulated debris and coolants, analysis and modeling, and auxiliary measurement of material properties and interactions. The specific power attained during the in-reactor experiment ranged up to about 1 kW/kg of UO_2 while the bulk sodium temperatures ranged up to 838 K. The particle-to-fluid interface does not appear to offer significant resistance to heat transfer at these power levels. The heat transfer behavior of the bed was quite reproducible, and edge effects appeared to be small.

- 2396 (NUREG/CR--0287) EXPERIMENTAL INVESTIGATION OF AEROSOL BEHAVIOR IN A POST-LMFBR ACCIDENT REACTOR CONTAINMENT ATMOSPHERE. Wegrzyn, J.; Shaw, D.T. (State Univ. of New York, Buffalo (USA). Dept. of Electrical Engineering). Jul 1978. 31p. (LAPES--78-02). NTIS \$4.50.

Results of experimental measurements of the aerodynamic properties of a sodium aerosol are presented. Essentially three separate experiments were performed; (1) the evaluation of the centrifuge as an in-situ measuring device, (2) the classification of the sodium aerosol, and (3) the shape factor measurements on an exploding wire (E.W.) aerosol.

- 2397 (NUREG/CR--0323(Vol.5)) ADVANCED REACTOR SAFETY RESEARCH. QUARTERLY REPORT, OCTOBER--DECEMBER 1977. (Sandia Labs., Albuquerque, NM (USA)). Oct 1978. Contract EY-76-C-04-0789. 134p. (SAND--78-0611(Vol.5)). Dep. NTIS, PC A07/MF A01.

Progress in LMFBR safety research is summarized for the following study areas: accident energetics, fuel dynamics, core debris behavior, aerosol source normalization, elevated temperature design assessment, accident delineation, and test and facility (ACPR Upgrade) technology.

- 2398 (NUREG/CR--0335(Vol.6)) ADVANCED REACTOR SAFETY RESEARCH. QUARTERLY REPORT, JANUARY--MARCH 1978. (Sandia Labs., Albuquerque, NM (USA)). Feb 1979. Contract EY-76-C-04-0789. 194p. (SAND--78-1421(Vol.6)). Dep. NTIS, PC A09/MF A01.

Progress is summarized in the following LMFBR safety-related areas: accident energetics, fuel dynamics, core debris behavior, aerosol source normalization, elevated temperature design assessment, accident delineation, and test and facility technology.

- 2399 (NUREG/CR--0339) LMFBR SAFETY. 6. REVIEW OF CURRENT ISSUES AND BIBLIOGRAPHY OF LITERATURE (1977). Buchanan, J.R.; Keilholtz, G.W. (Oak Ridge National Lab., TN (USA)). 13 Jul 1978. Contract W-7405-ENG-26. 105p. (ORNL/NUREG/NSIC--151). Dep. NTIS, PC A06/MF A01.

This report discusses the current status of liquid-metal fast breeder reactor (LMFBR) development. Selected bibliographic information on LMFBRs relative to the development and safety of the breeder reactor is presented for the year 1977. The bibliography consists of approximately 198 abstracts covering research and development, operating experience, and design practices. Keyword, author, and permuted-title indexes are included for completeness.

- 2400 (NUREG/CR--0340) LMFBR AEROSOL RELEASE AND TRANSPORT PROGRAM. QUARTERLY PROGRESS REPORT, APRIL--JUNE 1978. Kress, T.S.; Wright, A.L. (Oak Ridge National Lab., TN (USA)). 29 Sep 1978. Contract W-7405-ENG-26. 50p. (ORNL/NUREG/TM--244). Dep. NTIS, PC A03/MF A01.

Topics discussed include recent capacitor discharge vaporization tests in the CRI-III facility and the first shakedown test for the FAST facility; FAST facility installation progress; performance tests for the consumable electrode aerosol generator for the NSPP and comparison of results from three performance tests; performance of the first mixed sodium oxide and uranium oxide test in the NSPP; development of the backup plasma torch aerosol generator for use in generating high uranium oxide aerosol densities in the NSPP; and progress in development of a computer model to calculate the behavior of UO_2 vapor bubbles produced in core-disruptive accidents and in fuel-aerosol simulant test facility experiments.

- 2401 (NUREG/CR--0367) PROMPT BURST ENERGETICS EXPERIMENTS: FRESH OXIDE/SODIUM SERIES. Reil, K.O.; Young, M.F. (Sandia Labs., Albuquerque, NM (USA)). Aug 1978. Contract EY-77-C-01-0789. 83p. (SAND--78-1561). Dep. NTIS, PC A05/MF A01.

A series of in-pile experiments has been performed to provide information on thermal energy to work conversion under prompt burst excursion (PBE) conditions. These consisted of single pin tests using fresh uranium oxide or uranium carbide fuel in a capsule geometry, with either stagnant sodium or helium in the coolant channel. The experiments were irradiated with single or double pulses in the Annular Core Pulse Reactor (ACPR) to provide energy depositions up to 2900 J/g. This report covers the seven single and five double pulse UO_2 sodium-in tests. Experimental data includes pressure and linear motion transducer histories, measured work-energy conversion efficiencies, and post-irradiation examination. Analysis includes derived work-energy conversion efficiencies (up to 0.54%), pin failure modeling, hydrodynamic analysis of pressure pulse propagation in the channel, and piston stopping effects. Initial pressure events in the single pulse experiments appear to be dominated by fuel vapor pressure. Definite fuel-coolant interactions were observed in several experiments, including some that were coincident with stopping of the linear motion transducer piston, suggesting a possible triggering effect by the deceleration pressure.

- 2402 (NUREG/CR--0405) MARKOVIAN RELIABILITY ANALYSIS UNDER UNCERTAINTY WITH AN APPLICATION ON THE SHUTDOWN SYSTEM OF THE CLINCH RIVER BREEDER REACTOR. Papazoglou, I.A.; Gyftopoulos, E.P. (Brookhaven National Lab., Upton, NY (USA); Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). Sep 1978. Contract EY-76-C-02-0016. 258p. (BNL-NUREG--50864). Dep. NTIS, PC A12/MF A01.

A methodology for the assessment of the uncertainties about the reliability of nuclear reactor systems described by Markov models is developed, and the uncertainties about the probability of loss of coolable core geometry (LCG) of the Clinch River Breeder Reactor (CRBR) due to shutdown system failures, are assessed. Uncertainties are expressed by assuming the failure rates, the repair rates and all other input variables of reliability analysis as random variables, distributed according to known probability density functions (pdf). The pdf of the reliability is then calculated by the moment matching technique. Two methods have been employed for the determination of the moments of the reliability: the Monte Carlo simulation; and the Taylor-series expansion. These methods are adopted to Markovian problems and compared for accuracy and efficiency.

- 2403 (NUREG/CR--0408) NATURAL CONVECTION HEAT TRANSFER IN BEDS OF INDUCTIVELY HEATED PARTICLES. Rhee, S.J.; Dhir, V.K. (California Univ., Los Angeles (USA). School of Engineering and Applied Science). 1978. 93p. NTIS \$6.00.

Experimental observations of the onset of convective motion in beds of inductively heated particles have been made. The data for the heat transfer coefficient have been obtained in water-cooled beds of 304 stainless steel particles. The particulated bed was formed in a 10.4 cm diameter glass jar which was insulated on the sides and at the bottom. The free surface of the overlying layer of water was maintained at a constant temperature by a copper plate cooled with tap water. In the experiments, the height of the particulated bed and the height of the overlying fluid layer were varied systematically. The observations show that increasing the depth of the fluid layer over the bed tends to lower the critical internal Rayleigh number at which the onset of convection occurs.

- 2404 (NUREG/CR--0421) POSTACCIDENT HEAT REMOVAL: DEBRIS-BED EXPERIMENTS D-2 AND D-3. Rivard, J.B. (Sandia Labs., Albuquerque, NM (USA)). Aug 1978. Contract EY-76-C-04-0789. 57p. (SAND--78-1238). Dep. NTIS, PC A04/MF A01.

The second and third of the fission-heated experiments simulating fast-reactor debris in liquid sodium have been performed in the Annular Core Pulse Reactor (ACPR). These experiments were designed to assess the natural cooling capability that liquid sodium may provide for particulate debris which might be produced by a severe core-disruptive accident. Enriched UO_2 particles submersed in liquid sodium were used and were subjected to power levels ranging up to 1.28 kW/kg of oxide while the bulk sodium temperatures varied from 673 to 873 K. The particles were irregular in shape and ranged in size from about 100 to 1,000 μm . The D-2 bed loading was 600 kg/m² of oxide and the D-3 loading was 900 kg/m².

- 2405 (NUREG/CR--0425) AEROSOL MEASUREMENTS AND MODELING FOR FAST REACTOR SAFETY. QUARTERLY PROGRESS REPORT, JANUARY--MARCH 1978.

Gieseke, J.A.; Jordan, H.; Lee, K.W.;
Vaishnavi, B. (Battelle Columbus Labs., OH
(USA)). Sep 1978. 28p. (BMI--2005).
Dep. NTIS, PC A03/MF A01.

Technical progress during the quarter was made on the comparison of the HAARM-3 computer code with aerosol measurements, experimental measurements of uranium oxide aerosol characteristics, determinations of humidity effects on sodium aerosol formation, and development of the reference code. In addition, a mid-year review was presented to NRC staff on February 16, 1978. Technical progress on each of the above topics is discussed in the report.

- 2406 (NUREG/CR--0427) ACCIDENT PROGRESSION FOR A LOSS OF HEAT SINK WITH SCRAM IN A LMFB. Bari, R.A.; Ludewig, H.; Pratt, W.T.; Sun, Y.H. (Brookhaven National Lab., Upton, NY (USA)). Oct 1978. Contract EY-76-C-02-0016. 74p. (BNL-NUREG--50910). Dep. NTIS, PC A04/MF A01.

A description of a slow core meltdown in a liquid metal fast breeder reactor is presented for the conditions of loss-of-heat-sink following neutronic shutdown. Simple models are developed for the prediction of phase changes and/or relocation of the core materials including fuel, clad, ducts, control rod absorber material (B₄C), and plenum gases. The sequence of events is accounted for and the accident progression is described up to the point of recriticality. The neutronic behavior of the disrupted core is analyzed in R-Z geometry with a static transport theory code (TWOTRAN). For most scenarios assessed, the reactor is expected to become recritical although large ramp rates are not anticipated.

- 2407 (NUREG/CR--0443) CALCULATION OF SIGNALS IN CODED-APERTURE IMAGING OF MANY-PIN BUNDLES OF LMFB FUEL. McArthur, D.A.; Halbleib, J.A. Sr.; Morel, J.E. (Sandia Labs., Albuquerque, NM (USA)). Oct 1978. Contract EY-76-C-04-0789. 46p. (SAND--78-0503). Dep. NTIS, PC A03/MF A01.

Detailed Monte Carlo gamma-ray transport calculations have been performed to study the coded-aperture imaging problem for a 37-pin bundle. Realistic geometries have been used for the test fuel bundle, its containment, the coded aperture and the gamma-ray detector. A high-Z gamma-ray filter placed near the coded aperture was found to enhance the signal change caused by voiding of test fuel, and a non-linear relationship was found between the amount of fuel voided and the resulting signal change. The signal changes appear to be large enough that gamma-ray imaging would be useful for a 37-pin bundle.

- 2408 (NUREG/CR--0452(Vol.1)) USERS' MANUAL FOR THE SSC-L CODE. Agrawal, A.K. (Brookhaven National Lab., Upton, NY (USA)). Oct 1978. Contract EY-76-C-02-0016. 332p. (BNL-NUREG--50914(Vol.1)). Dep. NTIS, PC A15/MF A01.

A computer code, SSC-L, has been written on the CDC-7600 computing systems. This code simulates thermohydraulic transients in the entire liquid metal fast breeder reactor (LMFBR) plant. This report provides for all information related to the usage of this computer code to the potential users. It provides a complete description of modeling details and usage instructions with the exception of a generalized Plant Control and Protection Systems (PCS and PPS). These have also been modeled. These functions are needed to apply SSC-L for simulating operational transients.

- 2409 (NUREG/CR--0453(Vol.1)) SIMMER-II: A COMPUTER PROGRAM FOR LMFB DISRUPTED CORE ANALYSIS. Smith, L.L. (Los Alamos Scientific Lab., NM (USA)). Oct 1978. Contract W-7405-ENG-36. 279p. (LA--7515-M(Vol.1)). Dep. NTIS, PC A13/MF A01.

Physical models, and numerical methods are presented for SIMMER-II, a computer program to predict the neutronic and fluid dynamic behavior of an LMFB during a hypothetical core disruptive accident. Either the time-dependent multigroup neutron diffusion or neutron transport equation is solved by the quasistatic method to predict the reactor neutronic behavior during the accident. Cross sections depend on temperature and background cross sections. Three fields, a structure field, a liquid field, and a vapor field, are modeled to predict the fluid dynamic behavior of the reactor. Each field consists of density components, to follow the material motion, and energy components, to predict the material temperatures. For typical accident calculations, the materials are fertile fuel, fissile fuel, stainless steel, sodium, control material, and fission gas. Heat, mass and momentum transfer are calculated among the three fields and their components.

- 2410 (NUREG/CR--0453(Vol.2)) SIMMER-II: A COMPUTER PROGRAM FOR LMFB DISRUPTED CORE ANALYSIS. Smith, L.L. (Los Alamos Scientific Lab., NM (USA)). Sep 1978. Contract W-7405-ENG-36. 321p. (LA--7515-M(Vol.2)). Dep. NTIS, PC A14/MF A01.

Program description is presented for SIMMER-II, a computer program to predict the neutronic and fluid dynamic behavior of an LMFB during a hypothetical core disruptive accident. Either the time-dependent multigroup neutron diffusion or neutron transport equation is solved by the quasistatic method to predict the reactor neutronic behavior during the accident. Cross sections depend on temperature and background cross sections. Three fields, a structure field, a liquid field, and a vapor field, are modeled to predict the fluid dynamic behavior of the reactor. Each field consists of density components, to follow the material motion, and energy components, to predict the material temperatures. For typical accident calculations, the materials are fertile fuel, fissile fuel, stainless steel, sodium, control material, and fission gas. Heat, mass and momentum transfer are calculated among the three fields and their components.

- 2411 (NUREG/CR--0469(Vol.7)) ADVANCED REACTOR SAFETY RESEARCH QUARTERLY REPORT, APRIL-JUNE 1978. (Sandia Labs., Albuquerque, NM (USA)). May 1979. Contract EY-76-C-04-0789. 41p. (SAND--78-2001(Vol.7)). Dep. NTIS, PC A03/MF A01.

Progress in LMFB safety research is summarized for the following study areas: accident energetics; fuel dynamics; core debris behavior; aerosol source normalization; elevated temperature design assessment; LMFB accident delineation; and test and facility (ACPR upgrade) technology.

- 2412 (NUREG/CR--0470) ADVANCED REACTOR SAFETY RESEARCH. QUARTERLY REPORT, JULY-SEPTEMBER 1978. (Sandia Labs., Albuquerque, NM (USA)). Sep 1979. Contract EY-76-C-04-0789. 200p. (SAND--78-2002(Vol.8)). Dep. NTIS, PC A09/MF A01.

Progress in LMFB safety research is summarized for the following study areas: accident energetics, fuel dynamics, core debris

behavior, aerosol source normalization, elevated temperature-design assessment, accident delineation, and test and facility (ACPR upgrade) technology.

- 2413 (NUREG/CR--0477) PHYSICS OF REACTOR SAFETY. QUARTERLY REPORT, APRIL--JUNE 1978. (Argonne National Lab., IL (USA)). Aug 1978. Contract W-31-109-ENG-38. 91p. (ANL--78-85).

Work on LMFBR reactor safety programs, Monte Carlo analysis of safety-related critical assembly experiments, and reactor core thermal-hydraulic code development is summarized.

- 2414 (NUREG/CR--0482) LMFBR AEROSOL RELEASE AND TRANSPORT PROGRAM. QUARTERLY PROGRESS REPORT, JULY--SEPTEMBER 1978. Kress, T.S.; Wright, A.L. (Oak Ridge National Lab., TN (USA)). 10 Jan 1979. Contract W-7405-ENG-26. 44p. (ORNL/NUREG/TM--276). Dep. NTIS, PC A03/MF A01.

Progress is summarized for the Liquid-Metal Fast Breeder Reactor (LMFBR) Aerosol Release and Transport (ART) Program for the period July--September 1978. Program topics discussed include recent capacitor discharge vaporization (CDV) tests in the Fuel Aerosol Simulant Test (FAST)/Containment Research Installation (CRI-III) facility, including underwater and vacuum tests in the CRI-III vessel; analysis of data from the first Nuclear Safety Pilot Plant (NSPP) mixed-oxide aerosol test, performance and analysis of the second NSPP mixed-oxide aerosol test, and performance of the second NSPP sodium spray experiment; results from U_3O_8 aerosol tests in CRI-II using the dc plasma metal-oxygen torch, and development and testing of the dc plasma metal-oxygen torch for NSPP; and an evaluation of the HAARM-3 code fallout model using experimental data from NSPP tests.

- 2415 (NUREG/CR--0483) NUMERICAL RESULTS FOR A HEXAGONAL FUEL ASSEMBLY WITH A PLANAR BLOCKAGE USING THE COMMIX-1A COMPUTER CODE. Domanus, H.M.; Sha, W.T. (Argonne National Lab., IL (USA)). Sep 1978. Contract W-31-109-ENG-38. 136p. (ANL-CT--79-8). Dep. NTIS, PC A07/MF A01.

The numerical simulation of a 19-rod LMFBR fuel assembly with an edge blockage has been carried out using the computer code COMMIX-1A. This is a three-dimensional, transient, single-phase thermal-hydraulic computer code. The results have been obtained for high and low flow steady operating conditions. The computed outlet temperature distribution is in close agreement with the experimental measurements. The detailed results of velocity (axial and transverse) and temperature fields at various sections of the fuel assembly are presented to display the effects of planar blockage on the flow field.

- 2416 (NUREG/CR--0489) CHARACTERIZATION OF AGGLOMERATES OF UO_2 AEROSOL PARTICLES. Gordon, H.; Gieseke, J.A. (Battelle Columbus Labs., OH (USA)). Oct 1978. 29p. (BMI--2009). NTIS, PC A03/MF A01.

The report is the second report on a series of experiments designed to measure microscopic characteristics of aerosol particles that might be produced in core disruptive accidents of sodium cooled breeder reactors. The report covers experiments on UO_2 as a simulant of the radiotoxically most important aerosol species--the various isotopes of plutonium. Future experiments will be concerned with mixtures of these two species and with mixtures of fuel, coolant, and representative structural

material. For the branched chain-like particles characteristic of condensation aerosols of metallic oxides, two parameters have been found to be useful in describing deviation of particle conformation from the compact spherical: χ , a correction factor to the modified Stokes expression for fluid drag on a sphere and γ , a correction factor to the collision frequency for spherical particles undergoing coagulation. Results of experiments designed to measure the fluid drag correction factor χ are reported.

- 2417 (NUREG/CR--0527) SENSITIVITY ANALYSIS OF THE HAARM-3 CODE. Lee, K.W.; Gieseke, J.A.; Reed, L.D. (Battelle Columbus Labs., OH (USA)). Oct 1978. 92p. (BMI--2008). Dep. NTIS, PC A05/MF A01.

Battelle's Columbus Laboratories has recently developed and completed an improved computer model for predicting LMFBR aerosol behavior within the containment. This model is designated HAARM-3 (Heterogeneous Aerosol Agglomeration Revised Model-3). One of the important improvements incorporated in the HAARM-3 code compared to the previous models is that the effects of morphological properties of agglomerates (density, size, and shape) have been comprehensively corrected. Aerosol agglomeration mechanisms due to gas turbulence which had not been included previously has also been added into the HAARM-3 code. These HAARM-3 calculations have been compared with available experimental data and found to predict aerosol behavior more realistically. The report develops a methodology to perform sensitivity analysis of the HAARM-3 computer code and studies quantitatively the effects of various independent variables of the code such as initial aerosol properties and containment geometry upon aerosol behavior.

- 2418 (NUREG/CR--0587) AEROSOL MEASUREMENTS AND MODELING FOR FAST REACTOR SAFETY. QUARTERLY PROGRESS REPORT, APRIL 1--JUNE 30, 1978. Gieseke, J.A.; Jordan, H.; Schmidt, E.W.; Vaishnavi, B.; Lee, K.W. (Connecticut Yankee Atomic Power Co., East Hampton (USA)). Dec 1978. 36p. (BMI--2015). Dep. NTIS, PC A03/MF A01.

Technical progress during the quarter was made in areas of comparison of the HAARM-3 code with experimental data, sensitivity analysis of the code, reference code development, fuel aerosol agglomerate characteristics measurements, and evaluation of fuel aerosol resuspension. In addition, Battelle staff participated in a meeting at Oak Ridge National Laboratory on the 18th and 19th of April on the code verification plan.

- 2419 (NUREG/CR--0594) NUMERICAL THERMAL MIXING IN EULERIAN CALCULATIONS OF COUPLED THERMO/HYDRODYNAMIC MOTIONS. Sienicki, J.J.; Abramson, P.B. (Argonne National Lab., IL (USA)). Nov 1978. Contract W-31-109-ENG-38. 56p. (ANL--78-101). Dep. NTIS, PC A04/MF A01.

A great deal of effort has been devoted in the last five years by the reactor safety analysis community to the development of Eulerian computer codes for the combined thermodynamic and hydrodynamic analysis of accident progression. The systems to be analyzed have rather violent hydrodynamic motions initiated by strong pressure gradients which result from strong thermal gradients. Since hydrodynamic calculations in more than one dimension are easier to carry out in an Eulerian framework than in a Lagrangian framework, there has been strong motivation to

develop these new codes using Eulerian numerical schemes. Such schemes have a major drawback: they do not conserve thermodynamic potential energy. Eulerian schemes allow mass and energy to flow through calculational cells which are fixed in space. Pressures that drive the hydrodynamics are calculated from the average energy contents of the cells. Since mass and energy convect from cell to cell, at any time step the pressure of any cell represents the mass weighted average energy of that cell. Thus mass at higher specific energy is assumed by the nature of an Eulerian scheme to come instantaneously at each time step to thermal equilibrium with other mass in the cell at lower specific energy. It is shown that such numerical thermal mixing is unavoidable with pure Eulerian schemes and is a significant source of error in accident calculations.

- 2420 (NUREG/CR--0604) BUBBLE BEHAVIOR IN LMFBR CORE DISRUPTIVE ACCIDENTS. ANNUAL REPORT, OCTOBER 1977--SEPTEMBER 1978. Erdman, C.A.; Reynolds, A.B.; Chen, K.; Bradley, D.R.; Pilch, M. (Virginia Univ., Charlottesville (USA)). Dept. of Nuclear Engineering and Engineering Physics. Dec 1978. 120p. NTIS, PC A06/MF A01.

Analytical research performed at the University of Virginia during FY78 for the Reactor Safety Research (RSR) Division of the U.S. Nuclear Regulatory Commission is reported. The research is part of the Aerosol Release and Transport program of RSR. Three major areas were investigated: (1) particle production by hydrodynamic fragmentation, (2) particle production by homogeneous nucleation, and (3) bubble dynamics and heat transfer. Development of a Monte Carlo computer code for the analysis of particle-size distributions from vapor-liquid hydrodynamic fragmentation was initiated. The source of homogeneous nucleation in the ORNL CDV test was postulated to be expansion behind a spherical shock wave and modeling of this phenomenon was initiated. Analysis of the axial motion of bubble in the French EXCOBULLE tests was completed with the use of the EXCOBUL code in preparation for its use in assisting with the analysis of the ORNL under-water and under-sodium tests.

- 2421 (NUREG/CR--0620) NUMERICAL METHOD OF THE AEROSOL BEHAVIOR REFERENCE CODE, CRAB. Jordan, H.; Vaishnavi, B.; Gieseke, J.A. (Battelle Columbus Labs., OH (USA)). Jan 1979. 57p. (BMI--2016). NTIS, PC A04/MF A01.

A numerical method using arbitrary nodal spacing, linear elements and collocation at the nodal points for solving the aerosol agglomeration equation is described. A computer code employing this method has been developed and is named the CRAB (Contained Radioactive Aerosol Behavior) code. Comparisons between analytic solutions for two kernels, $K(x, x') = 1$ and $K(x, x') = x + x'$, and corresponding numerical calculations are given. In addition, a comparison run between the CRAB and two alternative codes, PARADISEKO-III and HAARM-3, for a lognormal initial distribution and a Brownian kernel is exhibited. An appendix traces the development of the present numerical method from related, but less suitable methods.

- 2422 (NUREG/CR--0653) PHYSICS OF REACTOR SAFETY. QUARTERLY REPORT, JULY--SEPTEMBER 1978. (Argonne National Lab., IL (USA)). Nov 1978. Contract W-31-109-ENG-38. 78p. (ANL--78-106). Dep. NTIS, PC A05/MF A01.

This report summarizes work done on reactor safety program by members of the Reactor Safety

Appraisals Group, Monte Carlo analysis of safety-related critical assembly experiments and Planning of Safety-Related (ZPR) Planning and Experiments Group. Work on reactor core thermal-hydraulic code development performed in the Components Technology Division is also included.

- 2423 (NUREG/CR--0676) AEROSOL MEASUREMENTS AND MODELING FOR FAST REACTOR SAFETY. ANNUAL REPORT FOR FY 1978. Gieseke, J.A.; Jordan, H.; Lee, K.W.; Reed, L.D. (Battelle Columbus Labs., OH (USA)). Jan 1979. 70p. (BMI--2021). Dep. NTIS, PC A05/MF A01.

The areas of major activity during that year were: development and subsequent sensitivity study of the HAARM-3 code; development of a code to study multiple zone effects; completion of the basic form of the reference code, CRAB; and measurements on fuel aerosol particles. Results of these activities are presented in the main body of this report. In addition, an appendix contains a compilation of experimental data on nuclear aerosols that have become available since the publication of a similar compilation in the previous annual report for this program.

- 2424 (NUREG/CR--0678) ANALYSIS OF THE HEAT AND MASS TRANSFER PROCESSES OF A UO_2 BUBBLE IN SODIUM FOR THE FUEL AEROSOL SIMULANT TEST (FAST). Tobias, M.L. (Oak Ridge National Lab., TN (USA)). 17 Apr 1979. Contract W-7405-ENG-26. 46p. (ORNL/NUREG/TN--307). Dep. NTIS, PC A03/MF A01.

The anticipated behavior of uranium oxide vapor bubbles produced by the capacitor discharge vaporization (CDV) method in the Fuel Aerosol Simulant Test (FAST) Facility is discussed on the basis of relatively simple physical models. Results of calculations for the rate of bubble rise and for heat and mass transfer rates are presented. Parametric studies indicate that future analysis efforts should emphasize the diffusion condensation process and the loss of heat from the bubble by radiation. Transfer of heat in the surrounding sodium is rapid enough that simplified models should be adequate. No important effects were noted in connection with bubble depth, initial quantity of UO_2 , or initial superheat.

- 2425 (NUREG/CR--0688) REACTOR SAFETY RESEARCH PROGRAMS. QUARTERLY PROGRESS REPORT, OCTOBER 1--DECEMBER 31, 1978. Agrawal, A.K.; Cerbone, R.J.; Jones, D.C. Jr.; Sastre, C.; Schweitzer, D.G.; van Rooyen, D. (Brookhaven National Lab., Upton, NY (USA)). Mar 1979. Contract EY-76-C-02-0016. 205p. (BNL-NUREG--50978). Dep. NTIS, PC A10/MF A01.

Current activities and technical progress are reported for the following projects: Gas Reactor Safety Evaluation; THOR Code Development; Code Review; SSC Code Development; LMFBR and LWR Safety Experiments; Fast Reactor Safety Code Validation; and Stress Corrosion Cracking of PWR Steam Generator Tubing.

- 2426 (NUREG/CR--0692) REPORTS DISTRIBUTED UNDER THE NRC REACTOR SAFETY RESEARCH FOREIGN TECHNICAL EXCHANGE PROGRAM. VOLUME VI, JULY--DECEMBER 1978. Queener, D.S. (Oak Ridge National Lab., TN (USA)). 6 Apr 1979. Contract W-7405-ENG-26. 111p. (ORNL/NUREG/NSIC--159; TID--3362(Vol.6)). Dep. NTIS, PC A06/MF A01.

Lists of documents exchanged during the second half of 1978 under agreements between the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research and the

governments of France, Federal Republic of Germany, Japan, and the United Kingdom are presented. These agreements cover safety research on high-temperature gas-cooled reactors, light-water reactors, and fast reactors. During this period, the NRC received 46 reports from France, 72 from the Federal Republic of Germany, 43 from Japan, and 93 from the United Kingdom. In return, the NRC sent 177 U.S. light-water reactor safety research reports to each of these four countries, 26 fast reactor safety research reports to all except France, and 7 HTGR research reports to Japan.

2427 (NUREG/CR--0721) LMFBR FUEL ANALYSIS. TASK A: OXIDE FUEL DYNAMICS. FINAL REPORT, OCTOBER 1977--SEPTEMBER 1978. Dhir, V.K.; Frank, M.; Kastenber, W.E.; McKone, T.E. (California Univ., Los Angeles (USA)). Dept. of Chemical, Nuclear, and Thermal Engineering). Mar 1979. 85p. NTIS, PC A05/MF A01.

Three aspects of LMFBR safety are discussed. The first concerns the potential reactivity effects of whole core fuel motion prior to pin failure in low ramp rate transient overpower accidents. The second concerns the effects of flow blockages following pin failure on the coolability of a core following an unprotected overpower transient. The third aspect concerns the safety related implications of using thorium based fuels in LMFBR's.

2428 (NUREG/CR--0736) LMFBR FUEL ANALYSIS. TASK C: RELIABILITY. Cave, L.; Kastenber, W.E. (California Univ., Los Angeles (USA)). School of Engineering and Applied Science). Nov 1978. 28p. NTIS, PC A03/MF A01.

The work done by UCLA in the form of consultative support on a variety of topics is summarized briefly. In the case of one particular topic, the problem of establishing an adequate verification program for natural circulation in FFTF, a probabilistic approach was developed which may be of more general interest in relation to other fast reactors. The UCLA paper on this topic is reproduced as an Appendix to this report.

2429 (NUREG/CR--0737) LMFBR FUEL ANALYSIS. TASK B. POST-ACCIDENT HEAT REMOVAL. FINAL REPORT, OCTOBER 1977--NOVEMBER 1978. Catton, I.; Dhir, V.K.; Brinsfield, W.A.; McDonough, J.M.; Ostrander, N.D. (California Univ., Los Angeles (USA)). School of Engineering and Applied Science). Nov 1978. 77p. NTIS, PC A05/MF A01.

This Task B Annual Report presents a review of PAHR problems and UCLA views upon areas that include debris bed phenomena, molten pool growth, gas film boundaries and crusts, and the influence of basemat conduction upon pool formation and evolution. Directed toward FFTF applications, the review introduces plausible circumstances for the formation of non-coolable debris beds, and so leads to crude estimates of the initial dimensions of core debris distributions for pool growth problems. The GROWS code is reviewed and examples given using initial conditions derived from those estimates.

2430 (NUREG/CR--0740) 1978 ANNUAL REPORT: AEROSOL CHARACTERIZATION FROM A SIMULATED HCDA. Zanotelli, W.A.; Miller, G.D.; Johnson, E.W. (Mound Facility, Miamisburg, OH (USA)). Mar 1979. 33p. (MLM--2597). NTIS, PC A03/MF A01.

Simulated environmental conditions modelling the HCDA on a reduced scale have provided the

following information: Aerosols resulting from the condensation of gaseous constituents are generally spherical, small in diameter (0.01 to 0.25 μm), and branched chain-like structures. Electron diffraction analyses have identified actinide dioxides, the constituents of stainless steel, and various sodium actinates (uranates and zirconates). Some superstoichiometric oxides (UO_2 and U_3O_8) were noted also. X-ray analyses were not sensitive enough to verify presence of sodium uranates or plutonates. Electron diffraction of the plutonia-doped system should verify these components in this mixed-actinide system. Reaction products are kinetically controlled during cool-down rather than based upon equilibrium thermodynamics.

2431 (NUREG/CR--0752) LMFBR AEROSOL RELEASE AND TRANSPORT PROGRAM. QUARTERLY PROGRESS REPORT, OCTOBER--DECEMBER 1978. Kress, I.S.; Tobias, M.L. (Oak Ridge National Lab., TN (USA)). Jul 1979. Contract W-7405-ENG-26. 45p. (ORNL/NUREG/TM--318). Dep. NTIS, PC A03/MF A01.

This report summarizes progress for the Liquid-Metal Fast Breeder Reactor (LMFBR) Aerosol Release and Transport (ART) Program sponsored by the Division of Reactor Safety Research of the Nuclear Regulatory Commission (NRC) for the period October--December 1978. This program is designed to investigate radionuclide release and transport from LMFBRs for reactor events of severity up to and including hypothetical core-disruptive accidents (HCDAs). Program topics discussed include recent capacitor discharge vaporization (CDV) tests in the Fuel Aerosol Simulant Test (FAST)/Containment Research Installation (CRI-III) facility, including underwater and argon tests in the CRI-III vessel; sodium spray fire experiments in the Nuclear Safety Pilot Plant (NSPP); results from U_3O_8 aerosol tests in CRI-II using the dc plasma metal-oxygen torch, and development and testing of the dc plasma metal-oxygen torch for NSPP; and a parametric study of heat and mass transfer mechanisms for a UO_2 bubble rising in sodium.

2432 (NUREG/CR--0759) PHYSICS OF REACTOR SAFETY. QUARTERLY REPORT, OCTOBER--DECEMBER 1978. (Argonne National Lab., IL (USA)). Mar 1979. Contract W-31-109-ENG-38. 68p. (ANL--79-19). Dep. NTIS, PC A04/MF A01.

Progress is summarized in reactor-safety research and technical coordination of the RSQ safety analysis program, Monte Carlo analysis of safety-related critical assembly experiments, planning of safety-related critical experiments, and work on reactor core thermal hydraulics.

2433 (NUREG/CR--0760) SIMMER-II ANALYSIS OF THE R-7 TREAT TEST. Bohl, W.P. (Los Alamos Scientific Lab., NM (USA)). Mar 1979. Contract W-7405-ENG-36. 30p. (LA--7763-MS). Dep. NTIS, PC A03/MF A01.

The SIMMER-II code was used in a simulation of the R-7 loss-of-flow TREAT test. Comparison with the test scenario encompassed thermocouple data, flowmeter tracers, hodoscope data, and the posttest configuration. Reasonably successful results were obtained on the overall heat balance. The calculated boiling time was within 0.2 s of the experimental result, and respectable agreement with the integrated lower flowmeter trace was achieved over the first 2 s of boiling. Although the details of cladding relocation and blockage formation could not be well simulated, the calculated fuel motion appears to be plausible given the available

experimental information. It was further concluded that this type of study is quite useful for revealing areas for model improvement.

- 2434 (NUREG/CR--0780) ANALYSIS OF AEROSOL PARTICLES UNDERGOING GRAVITATIONAL AGGLOMERATION. ANNUAL PROGRESS REPORT, OCTOBER 1, 1977--SEPTEMBER 30, 1978. Loyalka, S.K. (Missouri Univ., Columbia (USA). Coll. of Engineering). Feb 1979. 109p. NTIS, PC A06/MF A01.

Progress is summarized in development of appropriate theoretical expressions for the collision efficiency of LMFBR aerosol particles of arbitrary shape undergoing gravitational agglomeration and in experimental work to verify the theoretical expressions.

- 2435 (NUREG/CR--0785) COMMIX-1: A THREE-DIMENSIONAL TRANSIENT SINGLE-PHASE COMPONENT COMPUTER PROGRAM FOR THERMAL-HYDRAULIC ANALYSIS. Sha, W.T.; Domanus, H.M.; Schmitt, R.C.; Oras, J.J.; Lin, E.I.H. (Argonne National Lab., IL (USA)). Sep 1978. Contract W-31-109-ENG-38. 122p. (ANL--77-96). Dep. NTIS, PC A06/MF A01.

A mathematical model, COMMIX-1 Computer Program, for three-dimensional, transient, single-phase, thermal-hydraulic analysis of reactor components is presented. A set of governing equations for the conservation of mass, momentum, and energy is solved as a boundary-value problem in space and as an initial-value problem in time. The model can treat both continuum (e.g., reactor plenum) and quasi-continuum (e.g., rod bundles or fuel assembly) sodium systems. Volume porosity, surface permeability, and distributed resistance are used in the quasi-continuum approach, which alleviates two inherent deficiencies associated with conventional subchannel analysis; one is the approximation used in the formulation of transverse momentum equations, the other is its inability of handling reverse flow or recirculatory flow.

- 2436 (NUREG/CR--0790) TOPICAL REPORT ON FUEL MOTION DIAGNOSTIC SYSTEMS. Wright, S.A. (ed.). (Sandia Labs., Albuquerque, NM (USA)). Oct 1979. Contract EY-76-C-04-0789. 75p. (SAND--79-0909). Dep. NTIS, PC A04/MF A01.
- Five papers are presented concerning the performance of fuel motion detection systems for LMFBR type reactors and test facilities.

- 2437 (NUREG/CR--0790, pp 13-24) DEVELOPMENT OF A CODED APERTURE FUEL MOTION DIAGNOSTICS SYSTEM FOR THE ACPR (UPGRADE). Kelly, J.G.; Stalker, K.T. (Sandia Labs., Albuquerque, NM). Oct 1979.

Topical report on fuel motion diagnostic systems.

As part of Sandia Laboratories' program to study simulated core disruptive accidents in reactor safety research, a fuel motion detection system based on coded aperture imaging is being developed for the Annular Core Pulsed Reactor (ACPR). Although fuel motion has been observed at the TREAT by the fast neutron hodoscope and with a Vidicon pinhole camera technique, the coded aperture system offers a potential for lower cost, higher spatial resolution, three dimensional imaging, and higher frame rates at lower fluences than either of the other techniques.

- 2438 (NUREG/CR--0790, pp 25-38) ANALOG AND DIGITAL RECONSTRUCTIONS OF FUEL PIN IMAGES

RECORDED WITH THE PROTOTYPE ACPR (UPGRADE) FUEL MOTION DIAGNOSTIC SYSTEM. Stalker, K.T.; Kelly, J.G. (Sandia Labs., Albuquerque, NM). Oct 1979.

Topical report on fuel motion diagnostic systems.

A coded aperture imaging system to provide time resolved images of LMFBR fuel rods undergoing simulated core disruptive accidents is being developed at Sandia Laboratories. The imaging results of the SPR-II and ACPR experiments are presented. How the information was decoded from the coded image is discussed. The results of two different methods of image reconstruction are presented and the relative merits of both approaches are discussed.

- 2439 (NUREG/CR--0790, pp 39-50) POTENTIAL FOR USING CODED APERTURE IMAGING FOR FUEL MOTION DETECTION MEASUREMENT IN THE STF. McArthur, D.A.; Halbleib, J.A. Sr.; Morel, J.E.; Kelly, J.G. (Sandia Labs., Albuquerque, NM). Oct 1979.

Topical report on fuel motion diagnostic systems.

At Sandia Laboratories, coded aperture imaging of both neutrons and γ -rays is being investigated for fuel motion detection in STF as part of the NRC Reactor Safety Research program. Based on work done for the development of a fuel motion detection system for the ACPR UPGRADE, coded aperture imaging of γ -rays will be capable of producing high-resolution images of at least up to seven fuel pins. However, because of the short mean free path of γ -rays in the test fuel, it is not clear that γ -ray imaging can provide useful information in the large-bundle tests to be made in STF. Investigation of these questions in detail has therefore commenced using γ -ray transport codes. Alternatively, fast neutrons have a somewhat longer mean free path than γ -rays, and it may be desirable to image with fast neutrons in the case of larger fuel bundles.

- 2440 (NUREG/CR--0790, pp 51-62) RELATION BETWEEN FUEL MOTION AND DETECTOR RESPONSE FOR IN-CORE FUEL MOTION DETECTION SYSTEMS. McDaniel, P.J. (Sandia Labs., Albuquerque, NM); Wright, S.A.; Scott, W.H. Jr. Oct 1979.

Topical report on fuel motion diagnostic systems.

The detection of fuel motion within test sections of reactors which have no slots (SLSF, EBR-II, SCARABEE) requires the use of an in-core fuel motion detection system. In-core systems can also be useful as auxiliary fuel motion monitoring systems for reactor facilities which do have slots. A supplementary in-core fuel motion detection system would be especially useful for large-bundle tests, since self-shielding effects may degrade the resolution of external viewing systems. With support from the US Nuclear Regulatory Commission, an experimental program is being conducted at Sandia to demonstrate the feasibility of in-core fuel motion detection systems. This program has two major developmental efforts. The first requires development of a suitable set of detectors, and the second requires development of analytical methods which can determine fuel motion from detector signals.

- 2441 (NUREG/CR--0790, pp 63-74) FLASH X-RADIOGRAPHY FOR MATERIAL MOTION DETECTION. Choate, L.M.; Buckalew, W.H.; Posey, L.D. (Sandia Labs., Albuquerque, NM). Oct 1979.

Topical report on fuel motion diagnostic systems.

The applicability of flash x-ray

cinematography for the detection of material motion during reactor safety experiments is being studied at Sandia Laboratories. As part of this study, a comparison is being made between conventional radiographic and coded source radiographic techniques. For the conventional radiographic technique, the effects of photon buildup and photon scattering upon spatial and areal density resolution capabilities in poor geometry situations are being evaluated. Two particular methods of producing a coded radiographic source are under investigation.

- 2442 (NUREG/CR--0799) EXPERIMENTAL INVESTIGATION OF AEROSOL BEHAVIOR IN A POST-LMFBR ACCIDENT REACTOR CONTAINMENT ATMOSPHERE. Wegrzyn, J.; Shaw, D.T. (State Univ. of New York, Buffalo (USA). Lab. for Power and Environmental Studies). Mar 1979. 52p. NTIS, PC A04/MF A01.

During the past year the following four tasks have been completed: (1) the generation, handling and study of a dry sodium aerosol, (2) the comparison and evaluation of techniques for the determination of the various sodium compounds, (3) the calibration of the impactor and centrifuge for a pool fire sodium aerosol, and finally (4) the determination of the shape factor and effective density of a sodium pool fire aerosol.

- 2443 (NUREG/CR--0816) PRETEST PREDICTION OF THE W-1 SLSF EXPERIMENT USING THE COMMIX-1A COMPUTER CODE. Domanus, H.M.; Chen, M.J.; Sha, W.T. (Argonne National Lab., IL (USA)). May 1979. Contract W-31-109-ENG-38. 105p. (ANL-CT--79-33). Dep. NTIS, PC A06/MF A01.

The numerical simulation of a 19-rod LMFBR fuel assembly undergoing a Loss of Piping Integrity (LOPI) transient has been carried out using the COMMIX-1A computer code. COMMIX-1A is a three-dimensional, transient, single-phase, thermal-hydraulic computer code. The results are compiled to form a pre-test prediction of the W-1 in-pile experiment which will begin in the near future. Since COMMIX-1A performs single-phase analyses, the simulation is terminated when boiling first occurs in the fuel assembly.

- 2444 (NUREG/CR--0820) ADVANCED REACTOR SAFETY RESEARCH DIVISION. QUARTERLY PROGRESS REPORT, JANUARY 1--MARCH 31, 1979. Romano, A.J. (comp.). (Brookhaven National Lab., Upton, NY (USA)). May 1979. Contract EY-76-C-02-0016. 135p. (BNL-NUREG--51014). Dep. NTIS, PC A07/MF A01.

The Advanced Reactor Safety Research Programs Quarterly Progress Report describes current activities and technical progress in the programs sponsored by the USNRC Division of Reactor Safety Research. The projects reported each quarter are the following: HTGR Safety Evaluation, SSC Code Development, LMFBR Safety Experiments and Fast Reactor Safety Code Validation.

- 2445 (NUREG/CR--0839) STUDY OF THERMAL AND HYDRODYNAMIC PROCESSES ASSOCIATED WITH MELTING OF HORIZONTAL SUBSTRATE. Taghavi-Tafreshi, K.; Dhir, V.K. (California Univ., Los Angeles (USA). School of Engineering and Applied Science). May 1979. 83p. NTIS, PC A05/MF A01.

The melting of a horizontal slab of frozen olive oil placed beneath a pool of warm water has been studied experimentally. The interfacial heat flux data are taken in quasi-static mode by noting the time rate change of

enthalpy of the pool of water. Because of little agitation of the pool due to low melt volume flux ($\rho_{\text{melt}}/\rho_{\text{water}} \approx 1.09$; $\Delta T = 5$ to 45K), the pool was found to stratify with time. Hence, heat transfer coefficient data have been based on both the mean pool temperature and the interfacial temperature. Visual observations show that melt removal is governed by Taylor instability and that melt releasing nodes lie about a Taylor wavelength apart. Predictions of the interface growth based on equilibrium between surface tension and buoyant forces have been made and found to compare well with the data obtained from the movies. The heat transfer coefficient data obtained at higher pool temperatures are also predicted well by the theoretical model.

- 2446 (NUREG/CR--0844) LMFBR AEROSOL RELEASE AND TRANSPORT PROGRAM. QUARTERLY PROGRESS REPORT, JANUARY--MARCH 1979. Kress, T.S.; Han, J.T. (Oak Ridge National Lab., TN (USA)). Aug 1979. Contract W-7405-ENG-26. 56p. (ORNL/NUREG/TM--329). Dep. NTIS, PC A04/MF A01.

This report summarizes progress for the Liquid-Metal Fast Breeder Reactor (LMFBR) Aerosol Release and Transport (ART) Program sponsored by the Division of Reactor Safety Research of the Nuclear Regulatory Commission (NRC) for the period January--March 1979. This program is designed to investigate radionuclide release and transport from LMFBRs for reactor events of severity up to and including hypothetical core-disruptive accidents (HCDAs). Program topics discussed include recent capacitor discharge vaporization (CDV) tests in the Fuel Aerosol Simulant Test (FAST)/Containment Research Installation (CRI-III) facility, including underwater and vacuum tests in the CRI-III vessel; fuel-oxide tests in the Nuclear Safety Pilot Plant (NSPP) using a newly installed dc plasma metal-oxygen torch; fuel-oxide aerosol size measurements using two different instruments in the CRI-II facility, and calculated results of average size and number of primary particles per aerosol agglomerate; an evaluation of the HAARM-3 code fallout model using NSPP data; and the development of a simple analytical method to estimate the buoyancy-induced air velocity in the NSPP tests.

- 2447 (NUREG/CR--0846) ANALYSIS OF ORNL PIPE RATCHETTING TESTS TTT-1 AND TTT-2. Swenson, D.V. (Sandia Labs., Albuquerque, NM (USA)). Jul 1979. Contract EY-76-C-04-0789. 33p. (SAND--79-1057). Dep. NTIS, PC A03/MF A01.

The benchmark problem described is based on an experiment performed at the Oak Ridge National Laboratory (ORNL) to obtain data for qualification of high temperature inelastic analysis methods. Both the material and loadings are representative of conditions in a Liquid Metal Fast Breeder Reactor. In the experiment two straight pipe test specimens were subjected to thermal downshocks, changes in internal pressure, and creep at elevated temperature. These loadings produced progressive inelastic deformations (ratchetting) of the test specimens.

- 2448 (NUREG/CR--0864) PHYSICS OF REACTOR SAFETY. QUARTERLY REPORT, JANUARY--MARCH 1979. (Argonne National Lab., IL (USA)). May 1979. Contract W-31-109-ENG-38. 45p. (ANL--79-37). Dep. NTIS, PC A03/MF A01.

This Quarterly progress report summarizes work done during the months of January--March 1979 in Argonne National Laboratory's Applied Physics and Components Technology Divisions for

the Division of Reactor Safety Research of the US Regulatory Commission. The work in the Applied Physics Division includes reports on reactor-safety research and technical coordination of the RSR safety analysis program by members of the Reactor Safety Appraisals Group, Monte Carlo analysis of safety-related critical assembly experiments by members of the Reactor Computations Group, and planning of safety-related critical experiments by members of the Zero Power Reactor (ZPR) Planning and Experiments Group. Work on reactor core thermal-hydraulics was performed in ANL's Components Technology Division.

- 2449 (NUREG/CR--0872) CRAB USER'S MANUAL. Jordan, H.; Gieseke, J.A.; Schumacher, P.M. (Battelle Columbus Labs., OH (USA)). May 1979. 30p. (BMI--2027). NTIS, PC A03/MF A01.

The CRAB reference code for contained aerosol behavior is described. The model equation used is presented and the numerical solution technique -- finite element discretization coupled with the GEAR ordinary differential equation solver -- is discussed briefly. A detailed account of this technique is contained in an earlier report. Input required by CRAB is given in detail, as is the result of a sample run. CRAB is intended to provide a best estimate of aerosol behavior in a homogeneously mixed containment as a reference for less rigorous but more practical codes.

- 2450 (NUREG/CR--0884) UPDATED WORK PLAN FOR THE FAST-CRI-III FUEL VAPORIZATION AND TRANSPORT EXPERIMENTS. Wright, A.L.; Smith, A.M. (Oak Ridge National Lab., TN (USA)). Oct 1979. Contract W-7405-ENG-26. 39p. (ORNL/NUREG/TM--326). Dep. NTIS, PC A03/MF A01.

This report presents the work plan for the Containment Research Installation-III (CRI-III) tests and the Fuel Aerosol Simulant Tests (FAST) which constitute tasks 4 and 5, respectively, of the Aerosol Release and Transport Program, sponsored by the Division of Advanced Reactor Safety Research of the Nuclear Regulatory Commission. Both tests utilize the capacitor discharge vaporization technique to place uranium dioxide fuel samples into the high energy states that could be produced in Liquid-Metal Fast Breeder Reactor hypothetical core-disruptive accidents. The FAST experiments described will be performed under water and under sodium. These tests will help to develop models to predict fuel aerosol source release in accidents and also to verify models in the SIMMER accident analysis code being developed at Los Alamos Scientific Laboratory. The CRI-III tests will be performed to develop the capability to simultaneously vaporize three fuel samples; in addition, special testing will answer questions related to previous argon energy density tests.

- 2451 (NUREG/CR--0900) EVALUATION OF MATERIALS FOR RETENTION OF SODIUM AND CORE DEBRIS IN REACTOR SYSTEMS. ANNUAL PROGRESS REPORT, SEPTEMBER 1977-DECEMBER 1978. Swanson, D.G.; Zehms, E.H.; McClelland, J.D.; Meyer, R.A.; van Paassen, H.L.L. (Aerospace Corp., El Segundo, CA (USA). Materials Sciences Lab.). Dec 1978. 49p. NTIS.

This report considers some of the consequences of a hypothetical core disruptive accident in a nuclear reactor. The interactions expected between molten core debris, liquid sodium, and materials that might be employed in an ex-vessel sacrificial-bed or in the reactor

building are discussed. Experimental work performed for NRC by Sandia Laboratories and Hanford Engineering Development Laboratory on the interactions between liquid sodium and basalt concrete is reviewed. Studies of molten steel interactions with concrete at Sandia Laboratories and molten UO_2 interactions with concrete at The Aerospace Corporation are also discussed. The potential of MgO for use in core containment is discussed and refractory materials other than MgO are reviewed. Finally, results from earlier experiments with molten core debris and various materials performed at The Aerospace Corporation are presented.

- 2452 (NUREG/CR--0906) DYNAMIC INTRAGRANULAR FISSION GAS BEHAVIOR MODEL. Catton, I.; Kastenber, W.E. (California Univ., Los Angeles (USA). School of Engineering and Applied Science). Jul 1979. 214p. NTIS.

A model for the non-equilibrium behavior of intragranular fission gas in oxide fuel is developed to study the fundamental phenomena that determine fission gas effects in the fuel. The model explicitly represents the dynamic behavior of point defects and includes the effects of variations in stoichiometry. The mathematics of the model use the principle of moments invariance to allow approximations that significantly reduce computational expense without sacrificing accuracy. The model is intended for use as a tool to identify and explore important variables in the behavior of fission gas and determine the regimes of validity of the various modeling assumptions. The Dynamic Intragranular Gas Release and Swelling (DIGRAS) computer code developed in this work is applicable to simulations of both steady-state and transient irradiations. Flexibility is built into the code and a summary of the physical process dynamics is included in the code output.

- 2453 (NUREG/CR--0914) VISUAL INVESTIGATION OF REACTOR FUELS RESPONSE TO SIMULATED LOF HEATING CONDITIONS, FIRST SERIES. Cano, G.L.; Ostensen, R.W.; Young, M.F. (Sandia Labs., Albuquerque, NM (USA)). Oct 1979. Contract EY-76-C-04-0789. 76p. (SAND--79-0940). Dep. NTIS, PC A05/MF A01.

In a loss-of-flow (LOF) accident in an LMFBR, the mode and timescale of disruption of fuel can establish the probability of a subsequent energetic excursion. To investigate these phenomena, in-pile disruption of internal-fission-heated fuel pellets, both fresh UO_2 and preirradiated mixed oxide, was recorded by high speed cinematography. Neither fuel frothing nor dust-cloud breakup occurred under the simulated LOF conditions. Instead massive and very rapid fuel swelling, not predicted by current transient fuel-swelling and gas-release models, occurred. This may be the initial and dominant mode of fuel disruption in a LOF. A new transient fuel-swelling model, FISGAS, has been developed at Sandia.

- 2454 (NUREG/CR--0951) COMPARISON OF PIPING MODELS FOR DIGITAL POWER PLANT SIMULATORS. Sowers, G.W. (Arizona Univ., Tucson (USA). Dept. of Nuclear Engineering). Aug 1979. 45p. NTIS.

Thesis.
Two piping models intended for use in a digital power plant simulator are compared. One is a finite difference approximation to the partial differential equation called PIPE, and the other is a function subroutine that acts as a delay operator called PDELAY. The two models are compared with respect to accuracy and execution time. In addition, the stability of

the PIP3 model is determined. The PDELAY model is found to execute faster than the PIPE model with comparable accuracy.

- 2455 (NUREG/CR--0984(Vol.10)) ADVANCED REACTOR SAFETY RESEARCH. QUARTERLY REPORT, APRIL-JUNE 1979. (Sandia Labs., Albuquerque, NM (USA)). Nov 1979. Contract EY-76-C-04-0789. 228p. (SAND--79-1597(Vol.10)). Dep. NTIS, PC A11/MF A01.

Progress in LMFBR safety research is summarized for the following study areas: accident energetics; fuel dynamics; core debris behavior; aerosol source normalization; containment analysis; elevated temperature design assessment; LMFBR accident delineation; and test and facility technology.

- 2456 (NUREG/CR--1002) URANIUM AND SODIUM OXIDE AEROSOL EXPERIMENTS: NSPP TESTS 201-203 AND TESTS 301-302, DATA RECORD REPORT. Adams, R.E.; Kress, T.S.; Han, J.T.; Parsly, L.F. Jr. (Oak Ridge National Lab., TN (USA)). Nov 1979. Contract W-7405-ENG-26. 88p. (ORNL/NUR3G/TM--343). Dep. NTIS, PC A05/MF A01.

This data record report summarizes the first three uranium oxide aerosol tests and two preliminary mixed uranium oxide-sodium oxide aerosol tests conducted in the Nuclear Safety Pilot Plant project at Oak Ridge National Laboratory. The goal of the project is to establish the validity (or level of conservatism) of the aerosol behavioral code, HAARM-3, and follow-on codes under development at the Battelle Columbus Laboratories for the US Nuclear Regulatory Commission. Included in the report are descriptions of the five tests, together with tables and graphs summarizing the results.

- 2457 (NUREG/CR--1027) DISASSEMBLY PHASE ENERGETICS: AN EXAMINATION OF THE IMPACT OF SIMMER MODELS AND ASSUMPTIONS. Henninger, R.J.; Alcouffe, R.E. (Los Alamos Scientific Lab., NM (USA)). Sep 1979. Contract W-7405-ENG-36. 57p. (LA--7998-MS). Dep. NTIS, PC A04/MF A01.

We present an assessment of several aspects of material motion as predicted by the SIMMER code during the disassembly and early transition phase of an unprotected loss of flow transient in a liquid metal cooled advanced reactor. We divide the study into the fuel expansion phase of an energetic disassembly and into the early fuel motion stage of subassembly scale pin disruption under the forces of fission gas dispersal and sodium vapor flow. In the expansion phase we concentrate mainly on assessing the SIMMER estimates of energy partitioning of the expanding fuel bubble through the upper core and internal structures which may typically be present. Comparison with simple models leads us to conclude that the SIMMER formulation predicts a reasonable energy partition when heat transfer is not taken into account. The modeling of heat transfer is so complex that no general statement about the physical realizability of the SIMMER predicted energy partition due to the effect can now be made. This is particularly true when fuel to sodium heat transfer is assumed. We compared the subassembly scale early fuel motion with that given by the SLUMPY module in the SAS3A code. We find that SLUMPY may greatly overpredict the extent and magnitude of fuel motion in the time scales involved under LOF conditions using the nominal SLUMPY parameters.

- 2458 (NUREG/CR--1036) ADVANCED REACTOR

SAFETY RESEARCH DIVISION QUARTERLY PROGRESS REPORT, APRIL 1-JUNE 30, 1979. Agrawal, A.K.; Cerbone, R.J.; Jones, D.C. Jr.; Schweitzer, D.G. (Brookhaven National Lab., Upton, NY (USA)). Sep 1979. Contract EY-76-C-02-0016. 98p. Dep. NTIS, PC A05/MF A01.

Information is presented concerning HTGR fission product release and transport; properties of HTGR materials; LMFBR accident analysis; and SSC code development.

- 2459 (NUREG/CR--1058) IMPACT OF SIMMER-II MODEL UNCERTAINTIES ON PREDICTED POSTDISASSEMBLY DYNAMICS. Bell, C.R.; Burns, R.D. III; Luck, L.B. (Los Alamos Scientific Lab., NM (USA)). Oct 1979. Contract W-7405-ENG-36. 60p. (LA--8053-MS). Dep. NTIS, PC A04/MF A01.

The SIMMER-II calculation of the system kinetic energy developed during a voided core postdisassembly expansion in a hypothetical loss of flow (LOF) accident sequence gives results which are more than an order of magnitude below that for an ideal expansion. This work attempts to determine the extent to which uncertainties in the various phenomenological models in SIMMER-II influence the calculated results. A series of 15 separate SIMMER-II calculations were designed to make this assessment. The system kinetic energies which resulted ranged from 2.5 to 20 MJ compared to 100 MJ for the ideal expansion. It is concluded that experimental research in the area of voided core expansions should entail a balance between two areas. First, experiments should be performed that test the major uncertainties in the sequence of events as outlined in this report. Second, in order to increase confidence in the ability of SIMMER to predict other classes of expansion, testing of isolated phenomenological models should proceed.

- 2460 (NUREG/CR--1062) LMFBR AEROSOL RELEASE AND TRANSPORT PROGRAM. QUARTERLY PROGRESS REPORT, APRIL-JUNE 1979. Kress, T.S.; Wright, A.L. (Oak Ridge National Lab., TN (USA)). Jan 1980. Contract W-7405-ENG-26. 50p. (ORNL/NUREG/TM--354). Dep. NTIS, PC A03/MF A01.

This program is designed to investigate radionuclide release and transport from LMFBRs for reactor events of severity up to and including hypothetical core-disruptive accidents (HCDAs). Program topics discussed include recent capacitor discharge vaporization (CDV) tests in the Fuel Aerosol Simulant Test (FAST)/Containment Research Installation-III (CRI-III) facility, including underwater tests in the FAST vessel and underwater and vacuum tests in the CRI-III vessel; preliminary results from three Nuclear Safety Pilot Plant (NSPP) uranium oxide aerosol tests using the dc plasma metal-oxygen torch as an aerosol generator; results from CRI-III sodium oxide aerosol tests using the liquid-metal-spray sodium oxide aerosol generator; results from the normalization of the spiral-duct centrifuge calibration at different speeds; and the presentation of possible methods for calculating the rise in water temperature at the bubble vapor-liquid interface in FAST water experiments.

- 2461 (NUREG/CR--1084) AEROSOL MEASUREMENTS AND MODELING FOR FAST REACTOR SAFETY. QUARTERLY PROGRESS REPORT, OCTOBER 1-DECEMBER 31, 1978. Gieseke, J.A.; Lee, K.W.; Jordan, H. (Battelle Columbus Labs., OH (USA)). Oct 1979. 34p. (BNI--2024). NTIS. Technical progress during the quarter is

- described on the HAARM-3 computer code calculations and related subjects. THE HAARM-3 calculation results for the postulated accident as specified by the Committee on the Safety of Nuclear Installations are presented. Included also are the results of parametric studies with HAARM-3 on the geometry of Nuclear Safety Pilot Plant, and the progress in formulating a procedure for validating the HAARM-3 code.
- 2462 (NUREG/CR--1096) PHYSICS OF REACTOR SAFETY. QUARTERLY REPORT, APRIL-JUNE 1979. (Argonne National Lab., IL (USA)). Aug 1979. Contract W-31-109-ENG-38. 61p. (ANL--79-79). Dep. NTIS, PC A04/MF A01.
- This Quarterly progress report summarizes work done during the months of April-June 1979 in Argonne National Laboratory's Applied Physics and Components Technology Divisions for the Division of Reactor Safety Research of the US Nuclear Regulatory Commission. The work in the Applied Physics Division includes reports on reactor-safety research and technical coordination of the RSR safety analysis program by members of the Reactor Safety Appraisals Group, Monte Carlo analysis of safety-related critical assembly experiments by members of the Reactor Computations Group, and planning of safety-related critical experiments by members of the Zero Power Reactor (ZPR) Planning and Experiments Group. Work on reactor core thermal-hydraulics was performed in ANL's Components Technology Division.
- 2463 (NUREG/CR--1101) REPORTS DISTRIBUTED UNDER THE NRC REACTOR SAFETY RESEARCH FOREIGN TECHNICAL EXCHANGE PROGRAM. VOL. VII (JANUARY-JUNE 1979). Queener, D.S. (Oak Ridge National Lab., TN (USA)). Nov 1979. Contract W-7405-ENG-26. 91p. (ORNL/NUREG/NSIC--170; TID--3362(Vol.7)). Dep. NTIS, PC A05/MF A01.
- Lists of documents exchanged during the first half of 1979 under agreements between the US Nuclear Regulatory Commission's Office of Nuclear Regulatory Research and the governments of France, Federal Republic of Germany, Japan, and the United Kingdom are presented. These agreements cover safety research on high-temperature gas-cooled reactors (HTGR), light-water reactors, and fast reactors. During this period, the NRC received 43 reports from France, 81 from the Federal Republic of Germany, 17 from Japan, and 84 from the United Kingdom. In return, the NRC sent 163 United States light-water reactor safety research reports to each of these four countries, 50 fast reactor safety research reports to all except France, and 8 HTGR research reports to Japan.
- 2464 (NUREG/CR--1123) FUEL AEROSOL SIMULANT TEST DATA RECORD REPORT: ARGON TESTS. Wright, A.L.; Smith, A.M.; Rochelle, J.M.; Kress, T.S. (Oak Ridge National Lab., TN (USA)). Mar 1980. Contract W-7405-ENG-26. 67p. (ORNL/NUREG/TM--365). Dep. NTIS, PC A04/MF A01.
- This data record summarizes 13 uranium dioxide (UO_2) vaporization experiments performed in an argon gas environment in the Fuel Aerosol Simulant Test (FAST) project. The FAST project is part of the ORNL Aerosol Release and Transport Program sponsored by the Division of Reactor Safety Research of the Nuclear Regulatory Commission. The argon test series was performed to determine the influence of argon gas pressure and temperature on the mass of fuel vaporized and on the fuel aerosol size distribution produced for various amounts of energy injected into the fuel sample.
- Included in the report are descriptions of the 13 tests along with tables and graphs summarizing the results.
- 2465 (NUREG/CR--1124) FISGAS: A CODE FOR FISSION-GAS MIGRATION AND FUEL SWELLING IN AN LMFBR ACCIDENT. Ostensen, R.W. (Sandia Labs., Albuquerque, NM (USA)). Nov 1979. Contract EY-76-C-04-0789. 82p. (SAND--78-1790). Dep. NTIS, PC A05/MF A01.
- A new computer program, FISGAS, has been written to model the microscopic behavior of fission-gas in uranium dioxide reactor fuel and its effect on the macroscopic swelling of that fuel in a reactor accident. A radial multinode capability is provided. Separate calculations are provided for gas bubble motion and coalescence and gas transfer inside the grain, on the grain face, and on the grain edge. Microscopic modeling of the gas bubbles includes vacancy-flow restriction on the bubble size and macroscopic pressure field restraint on bubble growth. The pressure field is controlled by the fuel swelling rate which is limited by macroscopic creep. FISGAS is primarily designed to provide a diagnostic tool for the Fuel Disruption test series being performed in the Annular Core Research Reactor at Sandia Laboratories.
- 2466 (NUREG/CR--1134) POSSIBLE MECHANISMS FOR THE FORMATION OF THE OBSERVED AND EXPECTED COMPOUNDS DURING A HYPOTHETICAL CORE DISRUPTIVE ACCIDENT (HCDA). Zanotelli, W.A.; Miller, G.D. (Mound Facility, Miamisburg, OH (USA)). Oct 1979. Contract EY-76-C-04-0053. 23p. (MLM--2637). Dep. NTIS, PC A02/MF A01.
- Results are presented for an evaluation of the thermodynamics and kinetics of the possible chemical reactions which may occur in an HCDA bubble environment during its transport within, and following, its release from the reactor primary system.
- 2467 (NUREG/CR--1160) LMFBR SAFETY. 7. REVIEW OF CURRENT ISSUES AND BIBLIOGRAPHY OF LITERATURE (1978). Buchanan, J.R.; Keilholtz, G.W. (Oak Ridge National Lab., TN (USA)). Jan 1980. Contract W-7405-ENG-26. 85p. (ORNL/NUREG/NSIC--166; ORNL/NUREG/NSIC--125(Vol.7)). Dep. NTIS, PC A05/MF A01.
- This report discusses the current status of liquid-metal fast breeder reactor (LMFBR) development. Selected bibliographic information on LMFBRs relative to the development and safety of the breeder reactor is presented for the year 1978. The bibliography consists of approximately 127 abstracts covering research and development, operating experience, and design practices. Keyword, author, and permuted-title indexes are included for completeness.
- 2468 (NUREG/CR--1165) AEROSOL MEASUREMENTS AND MODELING FOR FAST REACTOR SAFETY. QUARTERLY PROGRESS REPORT, JANUARY 1-MARCH 31, 1979. Gieseke, J.A.; Jordan, H.; Lee, K.W. (Battelle Columbus Labs., OH (USA)). Nov 1979. 37p. NTIS.
- Technical progress during the quarter covered by this report was made in areas of the HAARM-3 calculations for postulated accidents, sensitivity analysis of the code, formulation of the HAARM-3 code validation procedure, and reference code development.
- 2469 (NUREG/CR--1213) ANS54: A COMPUTER SUBROUTINE FOR PREDICTING FISSION GAS RELEASE. Rausch, W.N.; Panisko, F.E. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jan

1980. Contract EY-76-C-06-1830. 24p. (PNL-3077). Dep. NTIS, PC A02/MF A01.

The ANS5.4 committee developed fission gas release models for stable and radioactive species for both low and high temperatures. This report reproduces these models and presents a FORTRAN coding of the models. The coding comprises subroutine ANS54 which is in a format suitable for direct insertion into a code like FRAPCON. Also shown are comparisons of model predictions with light water reactor low burnup data and liquid metal fast breeder reactor high burnup data that were used in the model development.

2470 (NUREG/CR--1252) PHYSICS OF REACTOR SAFETY. QUARTERLY REPORT, JULY-SEPTEMBER 1979. (Argonne National Lab., IL (USA)). Nov 1979. Contract W-31-109-ENG-38. 39p. (ANL--79-98). Dep. NTIS, PC A03/MF A01.

The work in the Applied Physics Division includes reports on reactor safety research and technical coordination of the RSR safety analysis program by members of the Reactor Safety Appraisals Group, Monte Carlo analysis of safety-related critical assembly experiments by members of the Reactor Computations Group, and planning of safety-related critical experiments by members of the Zero Power Reactor (ZPR) Planning and Experiments Group. Work on reactor core thermal-hydraulics was performed in ANL's Components Technology Division.

2471 (NUREG/CR--1285) COMPUTATIONAL RESULTS FOR A 7-PIN HEXAGONAL FUEL ASSEMBLY DURING A FLOW RUNDOWN TRANSIENT USING THE COMMIX-1A COMPUTER CODE. Domanus, H.M.; Chen, M.J.; Sha, W.T. (Argonne National Lab., IL (USA)). Jan 1980. Contract W-31-109-ENG-38. 52p. (ANL-CT--80-10). Dep. NTIS, PC A04/MF A01.

The numerical simulation of a 7-pin LMFBF fuel assembly undergoing flow rundown transients has been carried out using the COMMIX-1A computer code. This code performs three-dimensional, transient, single-phase thermal-hydraulic simulations. Transient results have been obtained for two different operating conditions. The computed transient temperatures compare favorably with the experimental measurements.

2472 (NUREG/CR--1294) MULTIPLE ZONE AEROSOL BEHAVIOR MODEL. Jordan, H.; Schumacher, P.M.; Gieseke, J.A.; Lee, K.W. (Battelle Columbus Labs., OH (USA)). Jan 1980. 37p. (BML--2042). NTIS.

The localized nature of the aerosol source of some LMFBF accident scenarios, such as a sodium pool fire in the containment of an LMFBF, contradicts a basic assumption of all presently employed nuclear aerosol behavior codes. This assumption was tested by creating a cellular model, the ZONE code, which accounts for spatial inhomogeneities in an approximate fashion. Calculations using ZONE and a homogeneous (QUICK code) model show that the homogeneous model gives consistently conservative results. This conservatism never exceeds a factor of about 2 for full scale reactor containments. Once the sodium fire extinguishes, the cellular and homogeneous models agree very closely.

2473 (NUREG/CR--1323) SIMULATION OF A THERMOHYDRAULIC TRANSIENT IN A PIPE USING THE COMMIX-1A COMPUTER CODE. Chen, M.J.; Domanus, H.M.; Sha, W.T. (Argonne National Lab., IL (USA)). Feb 1980. Contract W-31-109-ENG-38. 82p. (ANL-CT--80-15). Dep. NTIS, PC A05/MF

A01.

The numerical simulation of a section of a LMFBF intermediate pipe loop during a typical CRBR scram has been carried out using the COMMIX-1A computer code. COMMIX-1A is a three dimensional, transient, single-phase component computer code with both Cartesian and cylindrical coordinates for thermal hydraulic analysis. A considerable top to bottom temperature differences in the pipe is found as a result of flow stratification. The top to bottom temperature difference can be as high as approximately 50% of the temperature difference of the forcing function employed in the transient.

2474 (NYO--3930-11) LIQUID METAL BEARINGS TECHNOLOGY FOR LARGE, HIGH-TEMPERATURE, SODIUM ROTATING MACHINERY. DEVELOPMENT AND TEST OF HIGH-TEMPERATURE PROXIMITY PROBE FOR USE IN LIQUID SODIUM. FOURTH TOPICAL REPORT. Hoogenboom, I.; Kissinger, C. (Mechanical Technology, Inc., Latham, NY (USA)). Jun 1971. 148p. (MTI--71TR27). Dep. NTIS, PC A07/MF A01.

An inductance-type, proximity-probe system for measurement of bearing film thicknesses and shaft orbits in high-temperature, liquid-sodium environment has been developed, built and tested. The probe system is designed to operate at temperatures up to 1200°F and gap range of 0.004" to 0.080", with good linearity over about three-quarters of the range, when using probe pairs connected in push-pull. The probes were tested extensively in both gas and sodium environments, culminating in a 684 hours continuous test in a temperature range of 1050°F to 1170°F. The probes operated stably, repeatedly and without undergoing performance changes.

2475 (ORAU/IEA--79-2(M)) DOMESTIC AND INTERNATIONAL DEPLOYMENT OF REACTOR SYSTEMS: STATUS AND RATIONALE. Barkenbus, J.N.; Ohanian, M.J. (Oak Ridge Associated Universities, Inc., TN (USA). Inst. for Energy Analysis). Jan 1979. Contract EY-76-C-05-0033. 35p. Dep. NTIS, PC A03/MF A01.

The objectives of this report are to summarize the existing and projected world nuclear capacity by country and by reactor type, to examine the reasons for the ascendancy gained by light water reactors (LWRs), and to consider factors that might be relevant to the deployment of future nuclear energy systems.

2476 (ORNL--5062/R1) VENTURE: A CODE BLOCK FOR SOLVING MULTIGROUP NEUTRONICS PROBLEMS APPLYING THE FINITE-DIFFERENCE DIFFUSION-THEORY APPROXIMATION TO NEUTRON TRANSPORT, VERSION II. Vondy, D.R.; Fowler, T.B.; Cunningham, G.W. (Oak Ridge National Lab., TN (USA)). Nov 1977. Contract W-7405-ENG-26. vp. Dep. NTIS, PC A16/MF A01.

The report documents the computer code block VENTURE designed to solve multigroup neutronics problems with application of the finite-difference diffusion-theory approximation to neutron transport (or alternatively simple P_1) in up to three-dimensional geometry. It uses and generates interface data files adopted in the cooperative effort sponsored by the Reactor Physics Branch of the Division of Reactor Research and Development of the Energy Research and Development Administration. Several different data handling procedures have been incorporated to provide considerable flexibility; it is possible to solve a wide variety of problems on a variety of computer configurations relatively efficiently.

- 2477 (ORNL--5178) TSF MEASUREMENTS OF NEUTRON TRANSMISSION THROUGH AN AI-LMFBR LOWER AXIAL SHIELD MOCKUP. 189A NO. 10028, ACTIVITY NO. HN 04 01 01 1. Clifford, C.E.; Muckenthaler, F.J.; Stevens, P.N. (Oak Ridge National Lab., TN (USA)). Aug 1977. Contract W-7405-ENG-26. 128p. AT.

The results are presented of an experiment performed at the ORNL Tower Shielding Facility to measure neutron transmission through a mockup of a blanket=shield assembly designed by Atomics International for the lower axial region of the Liquid Metal Fast Breeder Reactor demonstration plant. The neutron source for the experiment was a collimated beam from the Tower Shielding Reactor II modified to simulate the energy spectrum of a fast reactor. The mockup contained a UO₂ blanket region and a sodium=flow transition region, both of which were penetrated by a cylindrical plug simulating various conditions in a central control=rod channel, plus upper and lower grid plates separated by a sodium plenum region. Measurements were made of the fluence and energy spectra of neutrons both within and emerging from several different configurations comprised of sections of the mockup. The complete set of data obtained in the experiment are presented.

- 2478 (ORNL--5181) ANALYSIS OF THE ZPR-9 REVERSE APPROACH TO CRITICAL (RAC) LOW LEVEL FLUX MONITOR EXPERIMENT. Selby, D.L.; Flanagan, G.F.; Fox, P.B.; Simpson, D.B. (Oak Ridge National Lab., TN (USA)). Jul 1979. Contract W-7405-ENG-26. 37p. AT.

An experiment was performed in the ZPR-9 facility at Argonne National Laboratory to determine the adequacy of the ex=core Low Level Flux Monitors (LLFMs) for monitoring the initial approach to critical in the Fast Flux Test Facility (FFTF). While the purpose of the LLFMs (which are located in the radial shield) is to monitor refueling operations, if they prove to be adequate for monitoring the initial loading operation they would be more convenient and less expensive than in=core detectors. From the analysis performed it was determined that the detectors were located so close to the core that spatial effects in the core became a dominant factor. The detector response was influenced more by localized events than by events occurring over the core as a whole. This made interpretation very difficult, if not impossible.

- 2479 (ORNL--5229/R1) INPUT DATA REQUIREMENTS FOR SPECIAL PROCESSORS IN THE COMPUTATION SYSTEM CONTAINING THE VENTURE NEUTRONICS CODE. Vondy, D.R.; Fowler, T.B.; Cunningham, G.W. (Oak Ridge National Lab., TN (USA)). Jul 1979. Contract W-7405-ENG-48. 51p. Dep. NTIS, PC A04/MF A01.

User input data requirements are presented for certain special processors in a nuclear reactor computation system. These processors generally read data in formatted form and generate binary interface data files. Some data processing is done to convert from the user oriented form to the interface file forms. The VENTURE diffusion theory neutronics code and other computation modules in this system use the interface data files which are generated.

- 2480 (ORNL--5255, pp 368-376) METALLURGICAL CHARACTERIZATION OF PROTOTYPIC TUBESHEET FORGING. Offer, H.P. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

Study of vacuum=arc=remelted 2 1/4 Cr=1 Mo steel tubesheet for the CRBR steam generator is reported. The microstructure resulting from heat treatment is shown; proeutectoid ferrite is the predominant phase. The flow pattern of the metal during closed=die, hot forging is represented schematically. Results of an inclusion content determination are tabulated. 3 figures, 1 table. (RWR)

- 2481 (ORNL--5255, pp 2-10) MECHANICAL PROPERTY PROGRAM OF ALLOY 718. Korth, G.E.; Smolik, G.R. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

Creep=rupture tests were performed on several heats given the conventional (1750°F) heat treatment. Elongation and time to rupture are tabulated for tests at 1200, 1350, and 1400°F; also rupture life is plotted vs stress. 0.2% yield stress, ultimate tensile strength, and elongation are tabulated for room=temperature tensile test data. Tests were conducted to determine which type of strain=controlled loading is the most damaging to alloy 718 in a creep=fatigue test. Results on stresses, cycles to failure, time to failure, etc., are tabulated. 2 figures, 4 tables. (RWR)

- 2482 (ORNL--5255, pp 11-28) LOW-CYCLE FATIGUE OF TYPE 16-8-2 STAINLESS STEEL WELD METAL. Raske, D.T. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

Data on the suitability of type 16-8-2 stainless steel as a filler metal for joining type 316 stainless steel in the fabrication of the CRBRP thermal liner were obtained. Continuous=cycling and 360=s tension hold=time fatigue tests on weld specimens are reported; tests were conducted in air at 593°C and a strain rate of 4×10^{-3} s⁻¹. Stresses and fatigue life are tabulated. Fatigue properties and cycle life were better than those of type 316 stainless steel. Monotonic 0.2% offset yield strength is higher in root weld metal than surface weld metal; however, the elastic moduli of samples from the two locations are the same. 8 figures, 2 tables. (RWR)

- 2483 (ORNL--5255, pp 28-36) APPLICATION OF DAMAGE=RATE APPROACH TO CREEP-FATIGUE LIFE PREDICTION ANALYSIS IN DIFFERENT HEATS OF TYPE 304 STAINLESS STEEL. Malya, P.S.; Majumdar, S. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

The damage=rate approach and its use are briefly summarized. Stress=relaxation data from hold=time experiments on three heats of type 304 stainless steel are tabulated. Calculated fatigue life is plotted vs the experimental value for tensile and compressive hold=time tests. Time to rupture is shown as a function of plastic strain rate. 4 figures, 1 table. (RWR)

- 2484 (ORNL--5255, pp 35-49) INTERPRETATION OF HIGH-TEMPERATURE CREEP OF TYPE 304 STAINLESS STEEL. Chopra, O.K.; Natesan, K. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

Second=stage creep behavior of type 304 stainless steel is evaluated in terms of a theory incorporating both dislocation=glide and recovery creep theories of steady=state creep.

- The applied stress dependence of the creep rate is examined in terms of the measured effective and internal stresses. The observed value of apparent activation energy is related to the true activation energy for the rate-controlling process for high-temperature creep. 9 figures, 2 tables. (RWR)
- 2485 (ORNL--5255, pp 53-56) INFLUENCE OF SODIUM ENVIRONMENT ON THE LOW-CYCLE FATIGUE BEHAVIOR OF AUSTENITIC STAINLESS STEEL. Smith, D.L.; Zeman, G.J. Mar 1977.
Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.
The creep-fatigue interaction of annealed type 304 stainless steel at 600°C was investigated. Low-frequency fatigue tests were conducted at a strain rate of $4 \times 10^{-5} \text{ s}^{-1}$ in sodium that contained 0.3 ppm carbon. For a total strain range of 1.1%, the fatigue lifetime was a factor of 2 lower than at $4 \times 10^{-3} \text{ s}^{-1}$. However, no effect of strain rate was observed at 0.94% strain range. Substantial differences in the plastic strain-range behavior were observed for the two rates. 2 figures. (RWR)
- 2486 (ORNL--5255, pp 58-64) FLOW AND FAILURE CRITERIA: INFLUENCE OF STRAIN RATE AND TEST TEMPERATURE ON DUCTILITY OF TYPES 304 AND 316 STAINLESS STEEL. Sikka, V.K.; Stephenson, R.L. Mar 1977.
Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.
Three-dimensional and isothermal plots are presented for uniform and total elongation and reduction of area for annealed type 316 stainless steel. The tensile strain rate varied over six orders of magnitude, and the test temperature ranged from 0 to 800°C. The observed quantities all showed high values, essentially independent of test temperature. 6 figures. (RWR)
- 2487 (ORNL--5255, pp 65-67) COMPLEX MATERIALS BEHAVIOR STUDIES IN SUPPORT OF DESIGN METHODS DATA LIST USING REFERENCE HEATS: COMPRESSIVE CREEP OF 16-MM (5/8-IN.) PLATE OF TYPE 316 STAINLESS STEEL REFERENCE HEAT. Sikka, V.K.; Baldwin, R.H. Mar 1977.
Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.
Problems in compressive creep testing are mentioned. An axial specimen of 0.25 in. diameter and 0.40 in. length was loaded to 15.5 ksi at 1100°F. Compressive strain-time data are plotted, and a photograph of the specimen after testing is shown. 3 figures. (RWR)
- 2488 (ORNL--5255, pp 67-90) MECHANICAL PROPERTY CHARACTERIZATION OF TYPE 316 STAINLESS STEEL REFERENCE HEAT. Sikka, V.K.; Baldwin, R.H.; Patton, E.B. Mar 1977.
Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.
Pipes, plates, and bars were reannealed at 1950°F for 0.5 h, then tensile tested at 77, 400, 800, 1100, and 1200°F at a strain rate of 0.0044/min. 0.2% yield stress, ultimate tensile strength, uniform and total elongation, and reduction of area are plotted, tabulated, and discussed for the various test temperatures. Reannealing lowered the yield strength, while ultimate tensile strength was not significantly affected. Data are also presented for each form from creep testing at 1100°F and 35.0 ksi; average time to rupture was 398 h. Tensile properties of 5/8-in. plates were measured for strain rates ranging from 0.22 to 0.00044 min⁻¹; temperature dependence of yield stress, ultimate tensile stress, elongation, and reduction of area are reported for the various strain rates. 18 figures, 5 tables. (RWR)
- 2489 (ORNL--5255, pp 90-97) ELASTIC PROPERTIES OF ENGINEERING CONSTRUCTION MATERIALS. Hammond, J.P. Mar 1977.
Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.
The dynamic elastic constants were determined for 9 Cr--1 Mo steel plate from 24 to 649°C (75 to 1200°F) by acoustical methods. Also, the static Young's moduli at room temperature were determined for mild carbon steel and Inconel 718 by use of a mechanical extensometer fitted with the LVDT strain sensors. The latter may be used as the 'X₀' values in the cubic polynomials for representing Young's modulus as a function of temperature for these alloys. Results for 9 Cr--1 Mo steel were compared with those for 2 1/4 Cr--1 Mo steel. 1 figure, 5 tables. (RWR)
- 2490 (ORNL--5255, pp 97-100) INCONEL ALLOY 718 MECHANICAL AND PHYSICAL PROPERTY PROGRAM. Roche, T.K.; Feltner, J.C. Mar 1977.
Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.
Creep-rupture tests were carried out from 1000 to 1400°F for stresses up to 150 ksi. Rupture time, elastic and plastic strains, and creep rate are tabulated. Temperature dependence of rupture properties was indicated. 1 figure, 1 table. (RWR)
- 2491 (ORNL--5255, pp 101-120) MECHANICAL AND METALLURGICAL BEHAVIOR OF WELDMENTS FOR LMFBR. Goodwin, G.M.; Brinkman, C.R. Mar 1977.
Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.
(I) Creep-rupture properties of type 16-8-2 stainless steel submerged-arc weld specimens with type 316 stainless steel base metal were measured. (II) Creep-rupture tests of CPF8 and CPF8M centrifugally cast pipe were performed at 1000 and 1200°F; data on strain, time to rupture, etc., are tabulated. Ductility at rupture and creep strain to rupture are plotted. (III) Tangential modulus of elasticity was determined for CPF8 and CPF8M piping, along with elastic compliance matrices. The results showed the importance of considering anisotropy in stress analyses for these materials. (IV) Variation of ultrasonic velocities with orientation in a multipass Inconel 82 weldment was measured. Both longitudinal and shear wave velocities vary by significant percentages in some planes. 14 figures, 5 tables. (RWR)
- 2492 (ORNL--5255, pp 130-135) HEAT-TO-HEAT VARIATION OF 2 1/4 CR--1 MO STEEL. Klueh, R.L. Mar 1977.
Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.
The 0.2% yield strength and ultimate tensile strength are plotted for 12 heats of 2 1/4 Cr--1 Mo steel in laboratory isothermally annealed condition. The dominant characteristic of the data is the dynamic strain aging peak when the ultimate tensile strength is given as a function of temperature. Much of the discussion

on the differences between heats is centered on dynamic strain aging and possible reasons for differences in behavior of the different heats. 1 figure, 1 table. (RWR)

- 2493 (ORNL--5255, pp 135-139) THERMAL AGING EFFECTS ON 2 1/4 CR--1 MO STEEL. Klueh, R.L. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

Room-temperature and aging-temperature tensile properties of aged commercial tubing heats of 2 1/4 Cr--1 Mo steel are tabulated for 850, 950, and 1050°F and aging times up to 5000 hours; properties included are yield, ultimate, and fracture strength, uniform and total elongation, and reduction of area. This material is to be used in the CRBR steam generators. 1 figure, 3 tables. (RWR)

- 2494 (ORNL--5255, pp 139-188) INTERIM ANALYSIS OF THE CREEP STRAIN--TIME CHARACTERISTICS OF ANNEALED AND ISOTHERMALLY ANNEALED 2 1/4 CR--1 MO STEEL. Booker, M.K. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

Constant-load isothermal creep strain--time data were analyzed over a range of stresses and temperatures to yield an expression for creep strain as a function of stress, time, and temperature. The equations contain terms involving elevated-temperature ultimate tensile strength, which greatly decrease the uncertainty in predictions caused by the large variations in the properties of the material due to heat treatment and heat-to-heat variations. This equation is recommended for use in the design of the CRBR steam generator. The analysis indicated that this material has two different creep mechanisms. 28 figures, 5 tables. (RWR)

- 2495 (ORNL--5255, pp 188-197) MECHANICAL PROPERTIES OF TRANSITION JOINT MATERIALS IN SUPPORT OF LMFR STEAM GENERATOR DESIGN. Brinkman, C.R.; King, J.F.; Strizak, J.P.; Klueh, R.L.; Booker, M.K. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

The area of primary concern requiring extensive characterization is the HAZ in the 2 1/4 Cr--1 Mo steel adjacent to the weld metal. Comparisons of the tensile properties obtained from annealed 2 1/4 Cr--1 Mo steel aged and tested at 510°C (950°F) indicate that aging reduces the ultimate tensile strength. The possibility of other aging effects at this temperature in the HAZ due to carbon depletion or precipitation of intermetallic phases or carbides needs to be investigated. Interim results of creep--rupture tests conducted on some of the various heat treatments (microstructures) possible across the HAZ are compared with similar all-weld-metal data. Annealed 2 1/4 Cr--1 Mo steel has the lowest creep--rupture strength but highest ductility at 510°C (950°F). Comparisons of interim individual material cyclic properties are given, including strain-controlled fatigue, cyclic stress--strain, and creep--fatigue interaction. The as-deposited ERNiCr-3 exhibited the highest cyclic strength and high-cycle fatigue resistance of the materials compared. 8 figures, 1 table. (RWR)

- 2496 (ORNL--5255, pp 197-204) CREEP AND

TENSILE PROPERTIES OF TRANSITION WELD JOINT MATERIALS. Klueh, R.L.; King, J.F. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

Joining the CRBR intermediate heat exchanger to the steam generator will require several joints, one of which will use Inconel 82 as a filler metal. Tensile properties of Inconel 82 weld metal (yield, ultimate, and fracture strengths, uniform and total elongations, and reduction of area) were determined for several plates at temperatures up to 1350°F. Analytical chemical results revealed that iron varied significantly across the cross section of the sample analyzed. The creep-rupture behavior of Inconel 82 weld metal is also briefly discussed. 4 figures, 2 tables. (RWR)

- 2497 (ORNL--5255, pp 205-222) HIGH-TEMPERATURE DESIGN. Corum, J.M.; Pugh, C.E. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

Uniaxial deformation studies on type 304 stainless steel emphasized nonsteady loading conditions including cyclic creep, reversal creep, and creep-plasticity interactions. Data obtained include tensile properties of specimens after discontinued creep, strain cycling behavior of specimens after discontinued creep, creep behavior of specimens after discontinued strain cycling, relaxation behavior during strain cycling, and creep behavior after interspersing of plasticity and creep strains of various types. Testing temperatures were 900, 1000, and especially 1100°F. Constant-strain-rate tests on type 316 stainless steel at 1100°F covered rates from 2.7×10^{-1} to 1.1×10^{-6} s⁻¹; stress--strain data are plotted. Cyclic hardening behavior of 316 SS at room temperature is exhibited, as are transient creep strain and plastic plus transient creep strain vs stress at 1100°F. Uniaxial and biaxial rupture data for type 304 stainless steel at 1100°F are compared. Axial and circumferential creep strains in constant-load biaxial rupture tests are shown. 22 figures. (RWR)

- 2498 (ORNL--5255, pp 270-277) FATIGUE CRACK PROPAGATION IN ALLOY 718: EFFECT OF HEAT TREATMENT ON MICROSTRUCTURE AND FATIGUE BEHAVIOR. Smith, H.H.; Michel, D.J. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

Fatigue crack propagation of Alloy 718 was investigated at 1100°F. Optical micrographs of microstructure and SEM examinations of fracture surfaces are presented. 3 figures, 2 tables. (RWR)

- 2499 (ORNL--5255, pp 279-287) DEFORMATION OF PRESSURIZED TUBES OF 316 SS AND ALLOY 718. Blackburn, L.D.; Steichen, J.M. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

Pressurized tube specimens of stainless steel 316 were tested at 900 to 1200°F at hoop stresses as high as 28.1 ksi. Corresponding numbers for Alloy 718 specimens were 700 to 1100°F and 77 ksi. Data for ambient temperature pressurization and hoop strain results for pressurized tubes after thermal exposure are tabulated. There is considerable scatter in the results. 5 tables. (RWR)

2500 (ORNL--5255, pp 287-300) **FATIGUE-CRACK PROPAGATION IN DISSIMILAR METAL WELDMENTS.** James, L.A. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

The crack propagation of three dissimilar metal welds finding application in LMFBR was studied. The specimens employed in this study were of the ASTM "Compact Type" design ($W = 2.00$ in., $B = 0.47$ in.), and were tested on a feedback-controlled machine with load as the control parameter. The cyclic frequency and stress ratio were 40 cpm and 0.05 for all tests. Tests were conducted at 427°C (800°F) and 538°C (1000°F) in an air environment. The three combinations of dissimilar metal weldments were Inconel 600/Inconel 82/Inconel 718, Inconel 718/Inconel 82/type 316 stainless steel, and Inconel 600/Inconel 82/type 316 stainless steel. An automatic gas-tungsten-arc (GTA) welding process was employed in welding the dissimilar base metals, using Inconel 82 filler wire in each case. The results for the three types of dissimilar metal weldments, each tested at two temperatures, are given. Also shown are scatter bands typical of the behavior of the base metals tested under identical conditions of temperature, frequency, loading waveform, and stress ratio. There is very little difference in the crack growth rates at the two temperatures. Fatigue-crack growth rates in the dissimilar metal weldments did not differ appreciably from those in the base metals tested under similar conditions. 6 figures. (RWR)

2501 (ORNL--5255, pp 302-314) **BASIC SPECIMEN TESTING.** Manjoine, M.J. (Westinghouse Research Labs., Pittsburgh). Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

Detailed data from creep testing of type 304 stainless steel (SS) are tabulated. Rupture data are shown for 1000°F, and creep rate is plotted for 1100°F. For a step-loaded test where stress is increased from a lower value after considerable exposure time, the creep rates are higher than for a continuous loading test at the higher stress level. Creep data for type 316 SS at 1100°F are also given. 2 figures, 2 tables. (RWR)

2502 (ORNL--5255, pp 314-318) **TUBULAR SPECIMEN TESTING.** Gowda, B.; Thomas, K.C. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

Most of the work concerned heat treatment of type 316 SS tubular specimens. Both heat-treated flat specimens and specimens cut from heat-treated tubular specimens were tested for tensile, hardness, and grain size properties. The heat treatment procedure with 30-minute soak time yielded the following properties: 0.2 percent yield stress, 28 to 32 ksi; ASTM Grain Size, 3 to 5; Rockwell B Hardness, 60 to 72. The variation of these three properties as a function of soak time is shown. The 30-minute soak time treatment yielded the desired properties. In addition, some testing was done on type 304 SS. 1 figure, 2 tables. (RWR)

2503 (ORNL--5255, pp 318-330) **SODIUM EFFECTS ON MECHANICAL PROPERTIES OF STRUCTURAL MATERIALS.** Flagella, P.N.; Kahrs, J.; Denne,

J.; Schrock, S.L. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

Results from creep rupture and low-cycle fatigue tests in flowing sodium of solution-annealed type 316 stainless steel with various pretreatments are presented. Tests were conducted at 593°C and stresses of 224.1, 248.2, and 275.8 MPa for Na exposures of 0, 2000, and 5000 hours. Results include strain on loading, hours to various percentages of creep strain, minimum creep rate, time to third-stage creep (0.2 percent offset), time to rupture, and total elongation at rupture. Effects of frequency and tension-hold on low-cycle fatigue life of type 316 stainless steel are shown. Stress range response and effects of strain rate and tension-hold on stress range response are also plotted. 7 figures, 2 tables. (RWR)

2504 (ORNL--5255, pp 377-392) **MECHANICAL PROPERTIES OF DECARBURIZED 2 1/4 CR--MO STEEL IN SODIUM.** Pizzo, P.P.; Hampton, L.V. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

The effect of decarburization of 2 1/4 Cr--1 Mo steel on its high-temperature mechanical properties was studied. Conditions relevant to CRBR operation were investigated. Uniaxial creep data are tabulated for 950°F testing after two different heat treatments. A thermal aging phenomenon is indicated. Biaxial creep data (time to rupture) were correlatable with uniaxial data; therefore, biaxial test results can be used to evaluate decarburization. Surface carburization was observed in a tube exposed to static sodium for 2000 hours at 950°F in a 304 SS exposure chamber. Carburization was also seen in type 316 stainless steel (SS) foils. To determine the decarburization potential of the loop sodium, foils of 316 SS and Cr--Mo steel were used; data indicated the decarburization of 2 1/4 Cr--1 Mo and the carburization of 316 SS. 5 figures, 3 tables. (RWR)

2505 (ORNL--5255, pp 393-410) **MECHANICAL PROPERTIES OF CARBURIZED STAINLESS STEEL IN SODIUM.** Copeland, J.F.; Pizzo, P.P.; Yuen, J.L. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

The influence of liquid sodium on fatigue crack propagation rates for annealed type 316 stainless steel and weldments for the LMFBR Intermediate Heat Transport System was studied. Testing was done at about 0.5 cycles per minute and 950°F in carbon-saturated sodium (to study carburization). Fatigue fracture surfaces of air and sodium specimens were examined with a scanning electron microscope. The carbon profiles of the crevice carburization exposure specimens were determined with an electron microprobe. Creep testing of carburized 316 H stainless steel continued; the time to rupture of the argon control specimen was approximately 2.6 times that of the carburized specimen, although creep rates were essentially equivalent. 7 figures, 3 tables. (RWR)

2506 (ORNL--5255, pp 411-423) **FRACTURE TOUGHNESS OF LMFBR STEAM GENERATOR MATERIALS.** Copeland, J.F.; Yuen, J.L. Mar 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending January 31, 1977.

Thick-section 2 1/4 Cr--1 Mo steel plates

were preheated to represent LMFBR conditions. Ten J-integral fracture toughness specimens were tested. Fracture toughness K_{Ic} values, as computed from J_{Ic} , illustrate the effects of temperature and welding on toughness. The elevated-temperature fracture toughness values for both the base metal and the shielded metal arc weld heat affected zone (HAZ) are significantly below the room-temperature results. Fracture toughnesses at room temperature are nearly identical for the HAZ and the base material. Results for gross membrane stress for fracture initiation are plotted as functions of crack size and fracture toughness. 7 figures, 2 tables.

- 2507 (ORNL--5281) HIGH-TEMPERATURE STRUCTURAL DESIGN PROGRAM. SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1976. Corum, J.M.; Pugh, C.E. (Oak Ridge National Lab., TN (USA)). Nov 1977. Contract W-7405-ENG-26. 236p. AT.

Progress in the development of high-temperature structural design methods for LMFBR components is reported under the following headings: constitutive equation development; inelastic structural tests and analyses; design criteria and rules for time-dependent deformation and failure; simplified design methods; thermal ratchetting tests; and structural design methods for weldments.

- 2508 (ORNL--5286) NONDESTRUCTIVE TESTING DEVELOPMENT PROGRAM QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1977. Hill, M.R. (Oak Ridge National Lab., TN (USA)). Jun 1977. Contract W-7405-ENG-26. 55p. AT.

Progress is reported for the following activities: (1) ultrasonic in-service inspection of EBR-II steam generator; (2) NDE support for departure nucleate boiling (DNB) test; (3) development of ultrasonic methods and equipment for LMFBR applications; (4) and advanced NDT development.

- 2509 (ORNL--5287) MECHANICAL PROPERTIES TEST DATA FOR STRUCTURAL MATERIALS. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING APRIL 30, 1977. Hill, M.R. (comp.). (Oak Ridge National Lab., TN (USA)). Jun 1977. Contract W-7405-ENG-26. 526p. AT.

This is the fifteenth in a series of progress reports, the purpose of which is to record and distribute quarterly the collected results of all structural materials mechanical properties test programs sponsored by the Reactor Development and Demonstration and the Nuclear Research and Applications Divisions of the Energy Research and Development Administration. Eight individual abstracts were prepared for the data base.

- 2510 (ORNL--5287, pp 17-86) ARGONNE NATIONAL LABORATORY. Weeks, R.W. Jun 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending April 30, 1977.

Results obtained on the fatigue behavior of Type 16-8-2 stainless steel weld metal, additional analysis of notch effects on the cyclic life of Type 304 stainless steel, and results from an experimental determination of bilinear stress-strain parameters for Incoloy 800 are described. An analysis of true stress-true strain tensile data obtained on titanium-modified Type 316 stainless steel exposed to flowing sodium is presented, and additional results on the effects of sodium environment on the fatigue behavior of Type 304 stainless steel are reported.

- 2511 (ORNL--5287, pp 87-353) OAK RIDGE NATIONAL LABORATORY. Slaughter, G.M. Jun 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending April 30, 1977.

The ORNL mechanical properties effort comprises several programs. The program Mechanical Properties and Behavior for Structural Materials deals with materials of general applicability to the LMFBR program. Other programs are concerned with the mechanical properties of weldments in LMFBR components, the specific materials of interest for LMFBR steam generators, collection and correlation of mechanical data needed for design on high-temperature LMFBR systems, and materials for the primary circuit and steam generators of HTGR systems. 160 figs., 51 tables, 123 refs.

- 2512 (ORNL--5287, pp 365-384) HANFORD ENGINEERING DEVELOPMENT LABORATORY. Yoshikawa, H.H. Jun 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending April 30, 1977.

The HEDL efforts involve evaluation of mechanical properties in the areas of normal and high strain rate tensile testing, creep testing and analysis, and fatigue crack growth and fracture toughness of such materials as weldments, vessel and piping, core support structures and reflectors as well as duct and cladding.

- 2513 (ORNL--5287, pp 385-424) WESTINGHOUSE ADVANCED REACTORS DIVISION. Bishop, E.C.; Ray, W.E. Jun 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending April 30, 1977.

Test programs are described involving basic specimen testing and tubular specimen testing of LMFBR type reactor materials. Sodium effects on mechanical properties of structural materials are also reported. (GHT)

- 2514 (ORNL--5287, pp 425-472) UNIVERSITY OF CINCINNATI. Motteff, J.; Jang, H. Jun 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending April 30, 1977.

The objective of this program is to (a) evaluate the time, temperature and stress-dependent mechanical properties of reactor structural materials, (b) determine the relationship of these properties to the microstructure, (c) show the contribution of the microstructure to the internal stress fields and the subsequent influence on microcracking and the grain boundary sliding behavior during the normal plastic deformation at elevated temperatures and (d) demonstrate the relationship of the hot micro-hardness properties with the hot-tensile properties of a class of commercial and advanced alloys presently under investigation at other laboratories. Special consideration is being given to operating conditions typical of nuclear reactor applications, including the knowledge that radiation environment can influence the substructure of these metals, a circumstance which can lead to significant changes in the conventional mechanical property behavior.

- 2515 (ORNL--5287, pp 474-485) METALLURGICAL

- CHARACTERIZATION OF PROTOTYPIC TUBESHEET FORGING. Offer, H.P. Jun 1977.
Mechanical properties test data for structural materials. Quarterly progress report for period ending April 30, 1977.
Room and elevated temperature tensile tests have been performed on the prototype lower tubesheet forging procured by GE from Cameron Iron Works (CIW). These preliminary tensile tests were performed on tubesheet specimens given either a full anneal plus a 40 hour 727°C (1340°F) post weld heat treatment (PWHT) (designated heat treatment P), or a full anneal plus this PWHT plus a 1000 hour 510°C (950°F) thermal aging/embrittlement treatment (designated heat treatment Q). The results are given in Table 8.2.1 along with the room temperature results supplied by CIW for heat treatment P.
- 2516 (ORNL--5287, pp 1-16) EG AND G IDAHO, INC. Keiser, D.D. Jun 1977.
Mechanical properties test data for structural materials. Quarterly progress report for period ending April 30, 1977.
The Alloy 718 Mechanical Property Program has as its primary objective the acquisition of the necessary mechanical property data to permit safe, efficient use of this material in LMFBR service. The specific immediate application is for selected structural components of the Clinch River Breeder Reactor Project (CRBRP). Other reactor service applications which require high levels of mechanical properties at temperatures up to 650°C (1200°F) are also expected to make use of this material. 4 tables, 7 figures.
- 2517 (ORNL--5293) SUBSTRUCTURES DEVELOPED DURING CREEP AND CYCLIC TESTS OF TYPE 304 STAINLESS STEEL (HEAT 9T2796). Swindeman, R.W.; Bhargava, R.K.; Sikka, V.K.; Moteff, J. (Oak Ridge National Lab., TN (USA)). Sep 1977. Contract W-7405-ENG-26. 36p. Dep. NTIS, PC A03/MF A01.
Substructures developed in tested specimens of a reference heat of type 304 stainless steel (heat 9T2796) are examined. Data include dislocation densities, cell and subgrain sizes, and carbide precipitate sizes. Testing conditions range for temperatures from 482 to 649°C, for stresses from 28 to 241 MPa, and for times from 4 to 15,000 hr. As expected, it is observed that temperature, stress, and time have strong influences on substructure. The change in the dislocation density is too small to measure for conditions which produce less than 1 percent monotonic strain. No cells form, and the major alteration of substructure is the precipitation of $M_{23}C_6$ carbides on grain boundaries, on twin boundaries, and on some dislocations. At stresses ranging from 69 to 172 MPa and at temperatures ranging from 482 to 593°C, the dislocation density increases with increasing stress and is generally higher than expected from studies made at higher temperatures. Dislocations are arranged in fine networks stabilized by carbides. At stresses above 172 MPa and temperatures to 649°C, the dislocation density is too great to measure. Cells develop which are finer in size than cells developed at similar stresses but at higher temperatures. Dislocation densities and cell sizes for cyclic specimens are comparable to data for creep-tested specimens. On the basis of the observed substructures, recommendations are made regarding further studies which would assist in the development of constitutive equations for high-temperature inelastic analysis of reactor components.
- 2518 (ORNL--5308, pp 250-253) CLINCH RIVER BREEDER REACTOR PLANT SAFETY DESIGN AND ELECTROMAGNETIC PULSE. Barnes, P.R. Jul 1978.
Health physics division annual progress report for period ending June 30, 1977.
The design of the Clinch River Breeder Reactor Plant is being examined to determine the probable effects of an intense electromagnetic pulse (EMP) environment. It has been found that the relatively high level of noise and electrical transient protection specified in the design will provide a high level of inherent hardness against an intense EMP environment. Thus, EMP is not considered a serious threat to the safety of the plant. However, as a precautionary measure, additional transient suppressors for some important battery power systems are recommended.
- 2519 (ORNL--5309, pp 261-265) NONDESTRUCTIVE TESTING. Smith, J.H. Nov 1977.
Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.
Development of an ultrasonic inspection technique for the weld and weld interfaces in dissimilar-metal transition joints is discussed. Work involving alloy 800, stainless steels 304 and 316, and 2 1/4 Cr--1 Mo is described. Frequency shifts were noted for some of the materials. 2 figures. (RWR)
- 2520 (ORNL--5309) LIQUID METAL FAST BREEDER REACTOR MATERIALS DEVELOPMENT PROGRAM. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1977. Peterson, S. (comp.). (Oak Ridge National Lab., TN (USA)). Nov 1977. Contract W-7405-ENG-26. 296p. AT.
LMFBR materials programs conducted or overseen by the ORNL Metals and Ceramics Division are reported. Major areas of interest include the following: steam generator materials development (steam-side/water-side corrosion--the effect of heat flux on the corrosion of 2 1/4 Cr--1 Mo steel by superheated steam, mechanical properties of steam generator materials, nondestructive testing), alternate LMFBR structural materials (advanced ferritic alloys development for LMFBR steam generators, alloy 800 development for advanced LMFBR steam generators), joining technology development (welding development and mechanical testing of weldments for stainless steel components, high-temperature structural design data for austenitic stainless steel weldments, hardfacing development for valves and bearings, basic welding and brazing research--examination of 16-8-2 simulated welds, advanced transition joints), piping and fitting development (including technology of transition joints), and advanced absorber materials (characterization of advanced materials, irradiation testing, performance analysis). Seventeen contributions with significant amounts of data are abstracted separately. (RWR)
- 2521 (ORNL--5309, pp 105-111) WELDING DEVELOPMENT AND MECHANICAL TESTING OF WELDMENTS FOR STAINLESS STEEL COMPONENTS. Edmonds, D.P.; Roche, T.K. Nov 1977.
Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.
304 and 316 stainless steels (SS) were welded by gas tungsten-arc and submerged arc processes with controlled-residual-element 308, 316, 16-8-2, and 17-8-2 SS used as the filler wire. Chemical compositions and ferrite numbers

are reported for these welds. Creep-rupture life and ductility at 1200°F are presented for 16-8-2 and 316 SS gas tungsten-arc welds. 2 figures, 4 tables. (RWR)

2522 (ORNL--5309, pp 111-114) HIGH-TEMPERATURE STRUCTURAL DESIGN DATA FOR AUSTENITIC STAINLESS STEEL WELDMENTS.

Goodwin, G.M.; Brinkman, C.R. Nov 1977.

Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.

Mechanical properties of 16-8-2 submerged-arc welds in type 316 stainless steel plate were evaluated. The welds were made under conditions similar to those used for the CRBR thermal liner. Room-temperature Charpy impact tests were carried out for 16-8-2 submerged-arc welds aged at 1050 and 1200°F for 1000 and 1500 hours. The trend appears to be a decrease in average impact strength with increased aging time. 1 figure, 1 table. (RWR)

2523 (ORNL--5309, pp 133-172) PIPING AND FITTINGS. McEnerney, J.W. Nov 1977.

Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.

Development of optimum fabrication methods for producing large-diameter (0.9 to 1.3 m), thin-walled (13 mm) austenitic stainless steel piping and fittings for LMFBR primary and secondary sodium loops is addressed. Fabrication methods were to be evaluated with respect to economic feasibility, dimensional control, and material behavior at LMFBR operating conditions. Formed-and-welded pipe and centrifugally cast-and-worked pipe are treated in considerable detail; dimensional evaluation was also performed on clamshell forged-and-welded elbows. 26 figures, 6 tables. (RWR)

2524 (ORNL--5309, pp 43-64) ANALYSIS OF THE TIME-DEPENDENT FATIGUE BEHAVIOR OF 2 1/4 CR--1 MO STEEL BASED ON THE CONCEPT OF LINEAR SUMMATION OF CREEP AND FATIGUE DAMAGE. Booker, M.K. Nov 1977.

Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.

Determination of parameters required by the model is considered first. The projected relation between strain range and cycles to failure at constant hold time is shown at 800, 900, 1000, and 1100°F; the relation between life to failure and hold period at constant strain range is also shown for the same temperatures. Elastic analysis design curves were constructed for strain range vs cycle life and cycles to failure. 24 figures, 1 table. (RWR)

2525 (ORNL--5309, pp 65-68) ANALYSIS OF THE FATIGUE CRACK GROWTH BEHAVIOR OF ISOTHERMALLY ANNEALED 2 1/4 CR--1 MO STEEL. Booker, M.K.; Corwin, W.R.; Booker, B.L.P. Nov 1977.

Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.

Attempts to develop analytical models to describe crack growth are summarized. Relations between crack growth rate and effective stress intensity factor, and crack growth constant and cycle period are shown. 2 figures. (RWR)

2526 (ORNL--5309, pp 77-95) ADVANCED FERRITIC ALLOYS DEVELOPMENT FOR LMFBR STEAM GENERATORS. Bryant, P.E.C. Nov 1977.

Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.

(I) Results of creep and stress rupture tests are tabulated and plotted. Deleterious effects of an Ar atmosphere were confirmed. (II) Plates were welded by the shielded metal-arc or gas tungsten-arc process. Hardness determined by Charpy V-notch impact tests is tabulated. 6 figures, 9 tables. (RWR)

2527 (ORNL--5309, pp 116-127) BASIC WELDING AND BRAZING RESEARCH: EXAMINATION OF 16-8-2 SIMULATED WELDS. Leitnaker, J.M.; Russell, K.F.; Houck, C.W.; Oliver, W. Nov 1977.

Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.

Results of Magne-Gage and hardness measurements of as-welded and aged weld metal are tabulated. Metallographic examinations revealed the presence of delta-ferrite in the starting materials. 7 figures, 1 table. (RWR)

2528 (ORNL--5309, pp 173-222) SURVEY OF AVAILABLE CREEP AND TENSILE DATA FOR ALLOY 800H. Booker, M.K.; Baylor, V.B.; Booker, B.L.P. Nov 1977.

Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.

Most of the transition joints of the CRBRP will consist of trimetallic joints employing a spool piece of alloy 800 H. A summary of available creep and tensile data for 800H is presented, including analytical representations. Strength properties, ductility properties, stress-strain relations, creep properties, rupture life, time to tertiary creep, creep strain to tertiary creep, and creep strain-time behavior are discussed. 35 figures, 8 tables. (RWR)

2529 (ORNL--5309, pp 223-245) CREEP AND TENSILE PROPERTIES OF TRANSITION WELD JOINT MATERIALS. Klueh, R.L.; King, J.F. Nov 1977.

Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.

Mechanical properties of filler materials used in welded transition joints of the CRBRP intermediate heat exchanger were investigated. Tensile properties, creep and rupture behavior of Inconel 82 (ERNiCr-3) welded metal are discussed. 13 figures, 5 tables. (RWR)

2530 (ORNL--5309, pp 245-261) FATIGUE PROPERTIES OF TRANSITION WELD JOINT MATERIALS. Strizak, J.P.; Brinkman, C.R.; King, J.F. Nov 1977.

Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.

CRBRP piping weld joints will use ERNiCr-3 (commonly known as Inconel 82) as a weld filler metal (among others). Fatigue data on ERNiCr-3 are presented. Tests conducted on material in the 2 1/4 Cr--1 Mo steel heat-affected zone adjacent the fusion line are also discussed. Cyclic stress-strain and continuous-cycle strain-controlled fatigue properties were defined over the range 22 to 5930C. A comparison between ERNiCr-3 deposited by the automatic hot- and cold-wire gas tungsten-arc processes indicated that fatigue life of hot-wire material was slightly superior to that of the cold-wire material; however, similar cyclic stress-strain response of the two materials was found. 12 figures, 3 tables. (RWR)

- 2531 (ORNL--5309, pp 7-22) CREEP AND RUPTURE BEHAVIOR OF 2 1/4 CR--1 MO STEEL. Klueh, C.R. Nov 1977.

Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.

Creep--rupture curves are shown for tempered bainitic steel plate and weld metal at 454, 510, and 566 C; a comparison of stress vs minimum creep rate relations is also given for the same conditions. Time and strain to onset of tertiary creep are tabulated for a 22-mm rod of normalized and tempered steel and for tempered weld metal. Data on stress to rupture in 10⁴ hours are given for bainitic and proeutectoid ferrite steels. 5 figures, 4 tables. (RWR)

- 2532 (ORNL--5309, pp 22-35) MECHANICAL PROPERTIES OF A 2 1/4 CR--1 MO STEEL FORGING FOR LMFBR TUBESHEETS. Klueh, R.L.; Griffith, J.L. Nov 1977.

Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.

Forging and heat treatments undergone by the specimen are recalled. Samples were taken at various positions and orientations. A definite change in microstructure along the radius was noted: the amount of bainite increased from 10% at the surface to 30% near the center of the cylindrical forging. Metallography revealed that two types of bainite formed. Tensile tests showed that an increasing amount of bainite led to increasing strength. Results for tensile properties are presented for various orientations, locations, and temperatures. The cooling process for the forging is discussed. It is felt that a more uniform microstructure is achievable by a normalizing treatment than by isothermal annealing. 7 figures, 4 tables. (RWR)

- 2533 (ORNL--5309, pp 1-7) STEAM-SIDE/WATER-SIDE CORROSION: THE EFFECT OF HEAT FLUX ON THE CORROSION OF 2 1/4 CR--1 MO STEEL BY SUPERHEATED STEAM. Griess, J.C.; DeVan, J.H.; Maxwell, W.A. Nov 1977.

Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.

Tests on vacuum-arc-remelted (VAR) steel are reported and compared with previous results from air-melted material. Specimen surface temperature ranged from 502 to 522°C. Data on weight change and penetration are presented. Effects of heat flux on penetration are noted. 3 figures, 2 tables. (RWR)

- 2534 (ORNL--5309, pp 36-38) THERMAL AGING EFFECTS ON 2 1/4 CR--1 MO STEEL. Klueh, R.L.; Griffith, J.L. Nov 1977.

Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.

Room-temperature and aging-temperature tensile properties are tabulated for aged commercial tubing heats of 2 1/4 Cr--1 Mo steel. Aging time runs to 10,000 hours. Temperatures are 850, 950, and 1050 F. 2 tables. (RWR)

- 2535 (ORNL--5309, pp 36, 39-43) EFFECT OF SODIUM ON THE MECHANICAL PROPERTIES OF 2 1/4 CR--1 MO STEEL. Klueh, R.L. Nov 1977.

Liquid Metal Fast Breeder Reactor Materials Development Program. Quarterly progress report for period ending March 31, 1977.

Creep rupture tests were conducted on

specimens that had been exposed to sodium for 26,500 hours at 1050°F; specimens aged in a helium atmosphere under similar conditions were also tested. Elongation as a function of rupture life went through a maximum for both the aged specimen and the one decarburized by sodium. The maximum was higher for the helium-aged material, but results appear similar for longer rupture life. All fractures were ductile transgranular fractures. 4 figures, 1 table. (RWR)

- 2536 (ORNL--5314) ANALYSIS OF ZPPR-5 SOURCE LEVEL FLUX MONITOR EXPERIMENTS. Selby, D.L.; Flanagan, G.F. (Oak Ridge National Lab., TN (USA)). Feb 1978. Contract W-7405-ENG-26. 56p. AT.

Analysis of 24 configurations of the ZPPR-5 Source Level Flux Monitor experiment has been performed with Modified Source Multiplication Techniques. Transport methods were used to calculate detection efficiencies for the Modified Source Multiplication equation. Detection efficiency changes as large as 40 percent were observed when going from near critical to approximately -34 (\$). Count rate ratios were calculated within 20 percent of the experimental values. Calculated-to-experimental reactivity ratios near 1.0 were calculated for both a ²³⁵U fission counter and a BF₃ detector.

- 2537 (ORNL--5346) MEASUREMENTS AND ANALYSIS OF THE CRBR INCONEL-STAINLESS STEEL RADIAL SHIELD EXPERIMENT. Maerker, R.E.; Muckenthaler, F.J. (Oak Ridge National Lab., TN (USA)). Jul 1978. Contract W-7405-ENG-26. 215p. AT.

Portions of document are illegible.

Neutron spectral measurements and Bonner ball counting rates were made behind similar thicknesses of inconel and stainless steel from 13 cm up to 39 cm to ascertain the better radial shield to an incident neutron spectrum similar to the one leaking the proposed CRBR. Calculations of the experiment were performed using a two-dimensional discrete ordinates code together with a standard design 51 group cross section set based on ENDF/B-IV. Both the calculations and the experiment indicated that inconel is superior to stainless steel, but the inconel cross sections are not as accurate in predicting the results of the experiment as are those of stainless steel.

- 2538 (ORNL--5349, pp 2-29) MECHANICAL PROPERTY PROGRAM OF ALLOY 718. Korth, G.E.; Smolik, G.R. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

Load-controlled high-cycle fatigue tests were conducted at 75, 800, 1000, 1100, and 1200°F; strain-controlled low-cycle tests were carried out at 75 and 1100°F. Data obtained are plotted; alloy 718 apparently exhibits an endurance limit. Results of Charpy V-notch impact tests for temperatures up to 1400°F are presented; the energy required for fracture seems to decrease as plate thickness increases. Stress--rupture data and data on tensile properties at 75 and 1200°F are tabulated for twin heats of alloy 718. Mechanical properties for a roll-extruded alloy 718 tube were investigated; microstructure (at 1750, 1830, 1900, and 2000°F), tensile properties (at 70, 650, 1200, and 1400°F), stress--rupture properties (at 1200 and 1300°F), and fatigue properties (at 1200°F) were determined. 8 figures, 13 tables. (RWR)

2539 (ORNL--5349, pp 31-43) LOW-CYCLE FATIGUE AND CYCLIC-DEFORMATION BEHAVIOR OF SMA TYPE 308 (CRE) STAINLESS STEEL WELD METAL AT 571°C. Raske, D.T. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

Continuous-cycling fatigue tests at a strain rate of $4 \times 10^{-3} \text{ s}^{-1}$ are reported for specimens machined longitudinally from various locations in a prototypic weld. Specimens were aged for $3.6 \times 10^4 \text{ s}$ at 593°C prior to testing. Fatigue and tensile data are tabulated, and total, elastic, and plastic strain ranges are plotted vs cycles to failure. Results are compared with similar ones for type 304 steel. The cyclic hardening and subsequent softening of the weld metal are shown as functions of strain range and cycles, along with the cyclic stress-strain curve. It was found that, for a given cyclic-strain range, the initial stress response is highly variable and remains so far at least half the fatigue life. 8 figures, 3 tables. (RWR)

2540 (ORNL--5349, pp 43-57) INFLUENCE OF SODIUM ENVIRONMENT ON THE LOW-CYCLE FATIGUE BEHAVIOR OF AUSTENITIC STAINLESS STEELS. Zeman, G.J.; Smith, D.L.; Natesan, K. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

Fatigue properties of solution-annealed, thermally aged, and sodium-exposed 304 and 316 stainless steel in sodium of controlled purity were investigated at 550, 600, and 700°C. The relation between plastic strain range and fatigue life for 304 stainless steel is shown for the three above conditions. Carburization of these specimens at 600°C had negligible effect on the fatigue life. At 550°C, however, the sodium-exposed specimens had a significantly smaller fatigue life than the aged material; the difference is attributed to carburization of the sodium-exposed specimens. To understand the effect of metallurgical history on the mechanical behavior under fatigue-loading conditions, hysteresis loops from different tests were analyzed, and plots of cyclic stress as a function of number of cycles were generated for specimens of 304 and 316 steels with different treatments. Optical photographs and scanning-electron micrographs of fracture regions of the two types of steel under different treatments are also shown. 8 figures. (RWR)

2541 (ORNL--5349, pp 60-64) COMPLEX MATERIALS BEHAVIOR STUDIES IN SUPPORT OF DESIGN METHODS DATA LIST USING REFERENCE HEATS. Sikka, V.K.; Baldwin, R.H.; Stevens, C.O. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

Tensile creep data at 593°C and 207 MPa are shown for 16-mm plates of type 316 stainless steel subjected to the following conditions: mill annealing, mill annealing and strain cycling, and reannealing and strain cycling; specimens were subjected to 10 and 1000 cycles. A comparison of tensile and compressive creep curves at 593°C and 107 MPa is made. 3 figures. (RWR)

2542 (ORNL--5349, pp 64-94) ANALYTICAL REPRESENTATION OF THE CREEP STRAIN--TIME BEHAVIOR OF COMMERCIAL HEAT-TREATED ALLOY 718. Booker, M.K. Dec 1977.

Mechanical properties test data for

structural materials. Quarterly progress report for period ending October 31, 1977.

Data were measured at temperatures from 538 to 760°C. Expressions for rupture life and temperature were obtained by generalized regression techniques; the time and creep strain to onset of tertiary creep were similarly analyzed. Relations between the various creep properties are presented. Normalized creep curves are plotted for a number of temperatures. Stress-relaxation and variable-load creep data are shown. 31 figures, 1 table. (RWR)

2543 (ORNL--5349, pp 94-101) MECHANICAL PROPERTIES OF 16-8-2 GIRTH WELDS ON LARGE-DIAMETER PIPE. Klueh, R.L.; Edmonds, D.P.; Bolling, E.W. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

Elevated-temperature properties of pipe welds prototypic of FFTF pipe welds were investigated. Short lengths of type 316 stainless steel pipe were welded by the gas tungsten-arc process. Tensile properties of longitudinal and transverse weld metal specimens are tabulated. 2 tables. (RWR)

2544 (ORNL--5349, pp 101-111) THERMAL AGING OF 2 1/4 CR--1 MO STEEL. Klueh, R.L. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

To study the possible effect of thermal exposure on mechanical properties of the steam generator material for the CRBR, material was aged at 454, 510, and 566°C for 1000, 2500, 5000, 10,000, and 20,000 hours. The effect of aging on creep is examined, and possible metallurgical processes leading to changes in creep and tensile behavior are discussed. The complex aging behavior is governed by complicated carbide precipitation processes; these are discussed at some length. Yield strength and ultimate tensile strength are compared as a function of aging time. 2 figures, 2 tables. (RWR)

2545 (ORNL--5349, pp 111-114) CONTINUOUS CYCLING AND CREEP--FATIGUE BEHAVIOR OF 2 1/4 CR--1 MO STEEL. Strizak, J.P. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

Results of continuous-cycling, strain-controlled fatigue tests at 427°C are tabulated, as are those from long-term creep--fatigue tests at 482°C. An interaction of tension and compression effects was suggested. Attempts were made to predict the number of cycles to failure. 2 tables. (RWR)

2546 (ORNL--5349, pp 114-136) MECHANICAL PROPERTY CHARACTERIZATION OF LARGE-DIAMETER PIPES OF TYPES 304 AND 316 STAINLESS STEELS. Sikka, V.K.; McEnerney, J.W. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

Tensile tests were completed on four formed-and-welded pipes of type 316 stainless steel; 0.2 percent yield strength, ultimate tensile strength, uniform elongation, and reduction of area are tabulated. Specimens tested were taken from both the base and weld (16-8-2 or 316) metal. The pipes were welded by different welding procedures. The tensile tests were supplemented by photographs of the specimens,

and the welds were macroetched to reflect their structure. A number of conclusions are drawn. In general, welds showed lower ultimate tensile strength and poor ductility in comparison with the base metal. Irrespective of the welding process and filler metal combinations tested, the weld was the weakest link in the formed-and-welded pipe. 15 figures, 4 tables. (RWR)

- 2547 (ORNL--5349, pp 137-146) TENSILE PROPERTIES OF ERNCR-3 (INCONEL 82) WELD METAL. Klueh, R.L.; King, J.F. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

Tests were conducted at 25 to 732°C and at strain rates of 3×10^{-6} and 3×10^{-4} s⁻¹. The following observations from these tests are discussed: differences in strength for welds made on 13-mm-thick plates (made with seven weld passes) and 19-mm-thick plates (made with about forty weld passes); isolated grain boundary cracks on voids; serrated yielding; and peaks in the yield strength and ultimate tensile strength as functions of temperature. An increase in flow stress at low strain rates between 500 and 650°C was tentatively attributed to an order effect. 3 figures. (RWR)

- 2548 (ORNL--5349, pp 146-171) CYCLIC STRESS-STRAIN BEHAVIOR OF ERNCR-3. Baylor, V.B.; Booker, M.K.; Strizak, J.P.; Klueh, R.L. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

Plates of annealed 2 1/4 Cr-1 Mo steel were welded by the gas tungsten-arc process with hot or cold wire ERNCR-3 filler metal. Longitudinal and transverse samples were taken for fatigue testing. Monotonic tensile curves obtained showed inconsistent slopes in the elastic portions of the curves; therefore, curves were reconstructed by a different procedure. At low strain ranges ERNCR-3 displayed rapid initial hardening followed by stable stress levels at all temperatures. At strain ranges above 1.94 percent, the hardening behavior was more complicated: hardening and softening behavior was observed. Cyclic hardening constants for cyclic stress-strain curves are tabulated. Neither the monotonic nor the cyclic bilinear stress-strain constants depended significantly on the type of weld filler metal used. In addition, the cyclic stress-strain response of ERNCR-3 did not appear to vary with the orientation of the specimen with respect to the line of fusion. From the bilinear representation of the monotonic data it appears that the bilinear slopes are independent of strain rate and temperature. 27 figures, 4 tables. (RWR)

- 2549 (ORNL--5349, pp 191-195) TENSILE PROPERTIES OF 304 SS AND 316 SS. Ward, A.L.; Blackburn, L.D. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

Tension test properties and flow stress values are tabulated for unirradiated materials to be used in the FFTF/CRBR Materials Irradiation Experiment. 3 tables. (RWR)

- 2550 (ORNL--5349, pp 196-199) HEAT-TO-HEAT AND/OR MELT PRACTICE VARIATIONS IN CRACK GROWTH BEHAVIOR OF INCONEL 718. James, L.A. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress

report for period ending October 31, 1977.

Fatigue-crack growth behavior for several heats of Inconel 718 is being studied at five temperatures. Results of room-temperature tests are presented. Metallographic examinations suggest that differences among the heats are due to the presence of networks of grain boundary carbides. 1 figure, 1 table. (RWR)

- 2551 (ORNL--5349, pp 199-200, 201-203) EFFECT OF STRESS RATIO UPON FATIGUE-CRACK GROWTH IN TWO STRUCTURAL ALLOYS AT ELEVATED TEMPERATURE. James, L.A. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

Fatigue-crack growth rates are expressed in a manner that accounts for the effect of stress ratio by use of the "effective stress intensity factor." Results are shown for SA-386 Grade 11 steel tested at 316 and 427°C and SA-637 Grade 688 (Inconel X-750) tested at 538°C. Each plot represents a fairly wide range of stress ratios, and k_{eff} provides a reasonably good correlation of growth rates over these ranges in stress ratio. 3 figures. (RWR)

- 2552 (ORNL--5349, pp 200, 204-211) EFFECT OF TEMPERATURE ON THE ELASTIC-PLASTIC FRACTURE TOUGHNESS BEHAVIOR OF INCONEL X-750. Mills, W.J.; Knecht, R.L. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

The $J_{\text{sub Ic}}$ fracture toughness behavior of heat-treated Inconel X-750 was characterized at 75, 800, and 1000°F. The material was tested in the double-aged, precipitation-hardened condition. Chemical composition and mechanical properties are tabulated. Fracture tests were performed on deeply cracked specimens. The critical value of J associated with crack initiation was determined by the R-curve technique of ASTM STP 514. The $J_{\text{sub Ic}}$ fracture toughness was relatively insensitive to temperature, although the slopes of the elevated-temperature fracture resistance curves were greater than the slope of the room-temperature R-curve. This phenomenon was associated with an intergranular to transgranular fracture mechanism transition. 4 figures, 2 tables. (RWR)

- 2553 (ORNL--5349, pp 232-245) SODIUM EFFECTS ON MECHANICAL PROPERTIES OF STRUCTURAL MATERIALS. Flagella, P.N.; Kahrs, J.; Denne, J.; Schrock, S.L. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

Results on creep-rupture and low-cycle fatigue tests in flowing sodium of solution-annealed type 316 stainless steel are described. Exposure to sodium for 5000 hours at 593°C before testing decreased creep resistance and increased ductility. Creep-rupture curves show the effects of stress and prior history. The material with higher creep rate showed greater ductility. Rupture life for a circumferentially notched bar was more than twice that of a bar with three transverse holes meeting in the center. Cyclic stress range response curves are presented for tests of 316 SS in flowing sodium at 593 and 538°C, with and without tension-hold time. In all cases the as-fabricated material yielded the highest stress range values, with stress range decreasing as pretreatment time increased. 11 figures. (RWR)

2554 (ORNL--5349, pp 264-268) FRACTURE TOUGHNESS OF LMFBR STEAM GENERATOR MATERIALS. Copeland, J.F. Dec 1977.

Mechanical properties test data for structural materials. Quarterly progress report for period ending October 31, 1977.

The G.E. program to evaluate the fracture toughness of 2 1/4 Cr-1 Mo is described briefly. Results on yield strength, ultimate tensile strength, elongation, and area reduction are quoted for 24, 399, and 510°C. 2 tables. (RWR)

2555 (ORNL--5350) NONDESTRUCTIVE TESTING DEVELOPMENT PROGRAM. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING SEPTEMBER 30, 1977. Hill, M.R. (comp.). (Oak Ridge National Lab., TN (USA)). Nov 1977. Contract W-7405-ENG-26. 58p. AT.

Nondestructive testing (NDT) research for LMFBR components is summarized. Included are NDT techniques and equipment for inspection and monitoring of welded joints and steam generators. NDT methods discussed employ ultrasonics, radiography, and eddy currents.

2556 (ORNL--5359) BREEDER REACTOR MATERIALS DEVELOPMENT PROGRAM QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JUNE 30, 1977. Brinkman, C.R.; Bryant, P.E.C.; Duke, J.M.; Goodwin, G.M.; King, R.T.; Tennery, V.J. (Oak Ridge National Lab., TN (USA)). Feb 1978. Contract W-7405-ENG-26. 235p. AT.

Progress is reported in steam generator materials, structural materials, joining technology, piping and fitting developments, and advanced absorber materials for breeder reactors.

2557 (ORNL--5366) BREEDER REACTOR MATERIALS DEVELOPMENT PROGRAM QUARTERLY PROGRESS REPORT FOR PERIOD ENDING SEPTEMBER 30, 1977. Patriarca, P.; Brinkman, C.R.; Bryant, P.E.C.; Duke, J.M.; Goodwin, G.M.; King, R.T.; Tennery, V.J. (Oak Ridge National Lab., TN (USA)). Apr 1978. Contract W-7405-ENG-26. 160p. AT.

Research is summarized on steam generator materials development, alternate breeder reactor structural materials, joining technology piping and fitting, and advanced absorber materials. (JRD)

2558 (ORNL--5369) LONG-RANGE PROGRAM PLAN FOR HIGH-TEMPERATURE STRUCTURAL DESIGN. (Oak Ridge National Lab., TN (USA)). Dec 1977. Contract W-7405-ENG-26. 101p. AT.

The program described is concerned with developing and validating the high-temperature structural design technology required for breeder reactor components and with the effective transfer of that technology to component designers. The primary objective of the program is thus to establish, and transfer to designers, a rationally sound and experimentally verified high-temperature structural design technology that will assure freedom from structural failure. More specific objectives are: (1) To provide a technology for assuring that component designs will be both safe and reliable during their design lifetimes. (2) To provide the designer with improved methods, procedures, tools, and criteria that are more economical to apply, are not overly conservative, and will allow increased component design flexibility, thus resulting in better and cheaper component designs and, ultimately, in increased flexibility in plant design and operation. (3) To provide methods for better definition of

design margins (against identified failure modes). (4) To experimentally verify and validate each aspect of the technology. (5) To improve reliability and confidence levels of component designs. (6) To provide design methods, and thus designs, that are more defensible. (7) To provide an improved technological basis for licensing.

2559 (ORNL--5378) MECHANICAL PROPERTIES OF TUBESHEET FORGINGS FOR LMFBR STEAM GENERATORS. Klueh, R.L.; Griffith, J.L. (Oak Ridge National Lab., TN (USA)). Apr 1978. Contract W-7405-ENG-26. 36p. AT.

Tensile and creep-rupture properties were determined on a 2 1/4 Cr-1 Mo steel forging that had been manufactured by techniques typical of those used for tubesheets for steam generators. Specimens were taken from a 0.64-m-dia by 0.30-m-thick cylindrical forging in the as-received condition and on small pieces of the forging given a laboratory isothermal anneal. The microstructures in both conditions were primarily proeutectoid ferrite with small precipitate particles throughout. The laboratory isothermally annealed steel contained about 10 percent bainite and minor amounts of pearlite. On the other hand, the microstructure of the as-received forging depended on the position of the forging from which the specimen was taken. Near the external cylindrical surface, the microstructure contained about 10 percent bainite. For specimens taken at intervals toward the center of the forging, the amount of bainite increased and went through a maximum of about 30 percent, then decreased to about 25 percent at the center. The strength also went through a maximum with distance into the forging. This nonuniformity of microstructure and strength was attributed to a nonuniformity of cooling rate for the large forging. Detailed tensile and creep-rupture properties were determined for the as-received forging on specimens taken from about midway between the external cylindrical surface and the center, and these properties were compared with those for the laboratory isothermally annealed specimens. The tensile properties of the as-received steel exceeded those for the pieces of steel given the laboratory isothermal anneal, with the as-received steel showing a much greater dynamic strain aging effect. At high stresses, the creep-rupture properties of the as-received forging were greater than those of the steel that received the laboratory anneal.

2560 (ORNL--5381) TEST OF THE DIFFUSION THEORY DIFFERENCE EQUATIONS IN SLAB GEOMETRY (COMPUTER CODE VANCER). Vondy, D.R.; Fowler, T.B. (Oak Ridge National Lab., TN (USA)). Jun 1978. Contract W-7405-ENG-26. 40p. Dep. NTIS, PC A03/MF A01.

The difference equations for representing the diffusion theory approximation to neutron transport depend on the method of formulation. A parameterized set of equations were implemented for slab geometry with mesh points located at material intersections. Such formulations as the mesh edge with four nearest neighbors on planes are included by the equations, as well as the higher coupling of eight nearest neighbors on planes for such formulations as linear finite element. The formulations of primary interest were applied to two-dimensional problems, and to three-dimensional problems with coupling to nearest neighbors on planes. Solutions were obtained to multigroup problems over a wide range of problem type. Selected results are reported together with conclusions regarding the formulations.

- 2561 (ORNL--5385) NONDESTRUCTIVE TESTING DEVELOPMENT PROGRAM. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1977. Hill, M.R. (comp.). (Oak Ridge National Lab., TN (USA)). Mar 1978. Contract W-7405-ENG-26. 80p. AT.

This is the eleventh in a series of progress reports, the purpose of which is to record and distribute quarterly the collected results of all nondestructive testing development programs sponsored by the Division of Reactor Research and Technology of the Department of Energy for LMFBR components. Information is presented for research work at the following contractors: Argonne National Laboratory; General Electric Company; Hanford Engineering Development Laboratory; and Oak Ridge National Laboratory.

- 2562 (ORNL--5386) THERMAL RATCHETTING TESTS OF TYPE 304 STAINLESS STEEL PIPE: SPECIMENS TTT-1 AND TTT-2. Corum, J.M.; Battiste, R.L.; Sartory, W.K.; Smith, J.E.; Young, H.C. (Oak Ridge National Lab., TN (USA)). Jun 1978. Contract W-7405-ENG-26. 116p. AT.

The results of two high-temperature thermal ratchetting tests on straight sections of type 304 stainless steel pipe are presented, and the results are used to evaluate inelastic analysis predictions of the ratchetting behavior. The pipe specimens, each of which was machined from a well-characterized heat of material, were subjected to a series of thermal downshocks on their inner surface, followed by sustained periods under an internal pressure loading at a temperature of 593°C (1100°F). Testing was carried out in a special sodium test facility built for the purpose, and the outer-surface strain histories were measured and compared with inelastic analysis predictions obtained using a one-dimensional finite-element procedure based on inelastic analysis guidelines recommended for use in design of U.S. LMFBR components. The required mechanical property data for the analyses were obtained from specimens of the specific heat of material used for the pipe ratchetting specimens. For one of the two tests, the agreement between experiment and predictions was extremely good, while in the other case the predictions significantly overestimated the measured response.

- 2563 (ORNL--5389) PRELIMINARY EVALUATION OF ALTERNATIVE FUEL CYCLE OPTIONS UTILIZING FAST BREEDERS. Burns, T.J.; White, J.R. (Oak Ridge National Lab., TN (USA)). Oct 1978. Contract W-7405-ENG-26. 47p. Dep. NTIS, PC A03/MF A01.

Various "alternative" fuel cycles have been proposed to alleviate concerns about reactor fuel being diverted for the manufacture of nuclear weapons. This report addresses the impact of such fuel cycles on the performance of the classical (homogeneous) oxide-fueled Liquid Metal Fast Breeder Reactor. The primary fuel cycle analyzed is the $^{233}\text{U}/^{238}\text{U}/^{232}\text{Th}$ fuel cycle in which the fissile component ^{233}U is "denatured" with ^{238}U in order to restrict the use of reactor fuel as a source of weapons-usable material. The denatured reactor performance as a function of the fissile enrichment (% ^{233}U in U) is evaluated. The associated energy center concept in which dispersed denatured reactors are coupled to ^{233}U production reactors operating in such secure energy centers is described. The symbiotic system of dispersed/energy center reactors is analyzed both from the standpoint of energy growth supported and dispersed/centralized power production. Lastly, the

effects of proposed changes in the nuclear data on both the reactor performance and symbiotic system characteristics are addressed.

- 2564 (ORNL--5398(Exec.Summ.)) INTERIM ASSESSMENT OF THE DENATURED ^{233}U FUEL CYCLE: FEASIBILITY AND NONPROLIFERATION CHARACTERISTICS. Abbott, L.S.; Bartine, D.E.; Burns, T.J. (eds.). (Oak Ridge National Lab., TN (USA)). Dec 1979. Contract W-7405-ENG-26. 74p. Dep. NTIS, PC A04/MF A01.

A fuel cycle that employs ^{233}U denatured with ^{238}U and mixed with thorium fertile material is examined with respect to its proliferation-resistance characteristics and its technical and economic feasibility. The rationale for considering the denatured ^{233}U fuel cycle is presented, and the impact of the denatured fuel on the performance of Light-Water Reactors, Spectral-Shift-Controlled Reactors, Gas-Cooled Reactors, Heavy-Water Reactors, and Fast Breeder Reactors is discussed. The scope of the R, D and D programs to commercialize these reactors and their associated fuel cycles is also summarized and the resource requirements and economics of denatured ^{233}U cycles are compared to those of the conventional Pu/U cycle. In addition, several nuclear power systems that employ denatured ^{233}U fuel and are based on the energy center concept are evaluated.

- 2565 (ORNL--5404) CREEP AND CREEP-RUPTURE BEHAVIOR OF ERNCR-3 WELD METAL. Klueh, R.L.; King, J.F. (Oak Ridge National Lab., TN (USA)). Jun 1978. Contract W-7405-ENG-26. 46p. AT.

Creep and creep-rupture tests were made on AWS A5.14 Class ERNCR-3 weld metal, commonly known as Inconel 82. Tests were made over the range 454 to 732°C on specimens taken from welds made by the gas tungsten-arc process. The results fell into two categories: those for tests at 454, 510, and 566°C displayed different characteristics than those at 621, 677, and 732°C. At the lower temperatures the creep curves were characterized by a rapid transient, a long steady-state stage, and little tertiary creep. At the elevated temperatures the transient stage was not as rapid, the steady-state creep stage was shorter, and the tertiary stage was longer. The low-temperature creep-rupture curves and stress-minimum creep rate curves were also much flatter than those at the elevated temperatures. For certain tests at 454, 510, and 566°C the creep curves exhibited an "instantaneous elongation" or "strain burst" phenomenon, in which the creep curves contained strain jumps. At these same temperatures, several specimens failed prematurely. Strain bursts are postulated to be related to the formation of dislocation pileups in the matrix. When a localized pileup breaks through a barrier, a dislocation avalanche results and triggers neighboring pileups continuing into neighboring grains giving rise to the propagation of a deformation band along the gage length in a Luder's-type deformation. It was proposed that the premature failures were related to the strain burst phenomenon and some scattered "voids" or "cracks" that were observed in the as-welded microstructure.

- 2566 (ORNL--5417) NONDESTRUCTIVE TESTING DEVELOPMENT PROGRAM. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1978. Hill, M.R. (comp.). (Oak Ridge National Lab., TN (USA)). May 1978. Contract W-7405-ENG-26. 76p. AT.

Nondestructive testing (NDT) research

applied to in-service inspection and monitoring of LMFBR components is described. The following NDT methods are included: radiography, eddy currents, and ultrasonics. (DG)

- 2567 (ORNL--5419) HIGH-TEMPERATURE STRUCTURAL DESIGN: 1977 SUMMARY AND STATUS REPORT. (Oak Ridge National Lab., TN (USA)). Jul 1978. Contract W-7405-ENG-26. 89p. AT.

The High-Temperature Structural Design (HTSD) Program is concerned with developing and validating the HTSD technology required for breeder reactor components and with the effective transfer of that technology to component designers. The primary objective of the program is thus to establish, and transfer to designers, a rationally sound and experimentally verified HTSD technology that will assure freedom from structural failure. Particular emphasis is placed on those failure modes associated with high-temperature, time-dependent material effects--creep rupture, creep-fatigue interaction, creep-enhanced ratchetting, creep buckling, and environmental enhancement of low-temperature failure modes. The HTSD Program has the following tasks: (1) development, (2) validation, (3) application, and (4) piping. The report is intended to reflect the unified nature of the High-Temperature Structural Design Program by providing, in highlight form, a summary of what was undertaken and accomplished in each of the tasks in relation to previously identified needs and plans. The document is intended to provide a concise overview of the progress being made in establishing an adequate high-temperature structural design technology and a convenient listing of references to more detailed reports and papers in specific areas of interest.

- 2568 (ORNL--5430) THERMAL RATCHETTING TEST OF TYPE 316 STAINLESS STEEL SEAMLESS TO SEAM-WELDED PIPE: SPECIMEN TTT-4. Sartory, W.K.; Battiste, R.L.; Corum, J.M.; Young, H.C. (Oak Ridge National Lab., TN (USA)). Sep 1978. Contract W-7405-ENG-26. 101p. AT.

The results of a ratchetting test of a type 316 stainless steel pipe are presented and are compared with inelastic analysis predictions. The test specimen was a pipe consisting of two sections: a seamless section and a seam-welded section with two diametrically opposed welds with 16-8-2 stainless steel filler metal. In a special sodium test facility, the specimen was subjected to a series of thermal downshocks at the inner surface, followed by sustained periods of high-temperature operation under internal pressure. The outer surface strain history was measured for both sections and was compared with the prediction of a one-dimensional finite-element analysis for the seamless section. The inelastic analysis procedure was based on guidelines recommended for use in the design of U.S. liquid-metal fast-breeder reactor components. Mechanical properties required for the analysis were obtained from tests using a material of the same heat and product form as that used for the ratchetting specimen. The agreement obtained between analysis and experimental results was judged to be acceptable throughout the test.

- 2569 (ORNL--5433) HIGH-TEMPERATURE STRUCTURAL DESIGN PROGRAM SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING JUNE 30, 1978. Corum, J.M.; Pugh, C.E. (Oak Ridge National Lab., TN (USA)). Nov 1978. Contract W-7405-ENG-26. 276p. AT.

This report summarizes progress in the ORNL

High-Temperature Structural Design Program for January through June 1978. The technical activities of the program are comprehensively reported under six major headings, or subtasks, which correspond to the subtasks of the current 189a OH048 document: constitutive equations for inelastic design analysis; inelastic structural tests and analyses; design criteria and rules for time-dependent deformation and failure; simplified design methods; thermal ratchetting tests; and structural design methods for weldments.

- 2570 (ORNL--5440) NONDESTRUCTIVE TESTING DEVELOPMENT PROGRAM. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JUNE 30, 1978. Hill, M.R. (comp.). (Oak Ridge National Lab., TN (USA)). Sep 1978. Contract W-7405-ENG-26. 89p. AT.

Nondestructive testing (NDT) research related to LMFBR materials and components testing is reported. Included are NDT techniques and equipment for inspection and monitoring of steam generator tubes and welded joints, sodium rupture discs, and DNB water tubes. NDT methods employing ultrasonics, radiography, and eddy currents are discussed.

- 2571 (ORNL--5466) RODMOD: A CODE FOR CONTROL ROD POSITIONING. Vondy, D.R.; Fowler, T.B. (Oak Ridge National Lab., TN (USA)). Nov 1978. Contract W-7405-ENG-26. 133p. Dep. NTIS, PC A07/MF A01.

The report documents a computer code which has been implemented to position control rods according to a prescribed schedule during the calculation of a reactor history. Control rods may be represented explicitly with or without internal black absorber conditions in selected energy groups, or fractional insertion may be done, or both, in a problem. There is provision for control rod follower, movement of materials through a series of zones in a closed loop, and shutdown rod insertion and subsequent removal to allow the reactor history calculation to be continued. This code is incorporated in the system containing the VENTURE diffusion theory neutronics and the BURNER exposure codes for routine use. The implemented automated procedures cause the prescribed control rod insertion schedule to be applied without the access of additional user input data during the calculation of a reactor operating history.

- 2572 (ORNL--5474) HIGH-RESOLUTION BORESIDE RADIOGRAPHY OF SMALL-DIAMETER TUBE-TO-TUBESHEET WELDS. Foster, B.E.; Davis, E.V.; McClung, R.W. (Oak Ridge National Lab., TN (USA)). Feb 1979. Contract W-7405-ENG-26. 43p. AT.

Nondestructive testing is essential to assure the integrity of the separation boundaries between water and sodium in the steam generators of the Liquid-Metal Fast Breeder Reactor. Radiography is required for the evaluation of the tube-to-tubesheet weld joints, the only welds in the tubing. A high-resolution boreside radiographic technique was developed to evaluate these welds. The method involves placing a very small-diameter radiation source in the bore of the tube in alignment with the weld joint, putting a radiographic film around the joint, and making a 360° panoramic exposure of the joint on a single radiograph. To develop the technique the authors: (1) studied film cassette design; (2) compared radiographic sensitivities between several isotopes (thulium, ytterbium, and iridium) and small-diameter (9-mm) microfocus rod-anode x-ray units; and (3) studied equivalent radiographic sensitivity of plaque-

and wire-type penetrameters. Radiographic sensitivities of approximately 1% were achieved with the microfocus x-ray unit, while the best sensitivity obtained with the isotopes was approximately 3%. The resolution and sensitivity achieved in boreside radiography of small-diameter tube-to-tubesheet welds with the microfocus x-ray unit are superior to those of any known technique. The technique has been successfully applied to welds in tubing with inner diameters ranging from 10 to 32 mm. Similar results should be obtainable in tubing of even larger diameter. The established boreside radiographic technique was used to monitor weld quality as a function of welding parameters during the weld development program for the Clinch River Breeder Reactor (CRBR) at Atomic International. The technique and equipment are to be used to evaluate the tube-to-tubesheet welds during the manufacturing phase of the CRBR steam generators.

- 2573 (ORNL--5477/V1) SACRD: A DATA BASE FOR FAST REACTOR SAFETY COMPUTER CODES, GENERAL DESCRIPTION. Greene, N.M.; Forsberg, V.M.; Raiford, G.B.; Arwood, J.W.; Simpson, D.B.; Flanagan, G.F. (Oak Ridge National Lab., TN (USA)). Jan 1979. Contract W-7405-ENG-26. 29p. Dep. NTIS, PC A03/MF A01.

SACRD is a data base of material properties and other handbook data needed in computer codes used for fast reactor safety studies. Data are available in the thermodynamics, heat transfer, fluid mechanics, structural mechanics, aerosol transport, meteorology, neutronics, and dosimetry areas. Tabular, graphical and parameterized data are provided in many cases. A general description of the SACRD system is presented in the report.

- 2574 (ORNL--5477/V2) SACRD: A DATA BASE FOR FAST REACTOR SAFETY COMPUTER CODES, CONTENTS AND GLOSSARY OF VERSION 1 OF THE SYSTEM. Greene, N.M.; Forsberg, V.M.; Raiford, G.B.; Arwood, J.W.; Flanagan, G.F. (Oak Ridge National Lab., TN (USA)). Jan 1979. Contract W-7405-ENG-26. 80p. Dep. NTIS, PC A05/MF A01.

SACRD is a data base of material properties and other handbook data needed in computer codes used for fast reactor safety studies. This document lists the contents of Version 1 and also serves as a glossary of terminology used in the data base. Data are available in the thermodynamics, heat transfer, fluid mechanics, structural mechanics, aerosol transport, meteorology, neutronics and dosimetry areas. Tabular, graphical and parameterized data are provided in many cases.

- 2575 (ORNL--5483) INSTRUMENTATION AND CONTROLS DIVISION BIENNIAL PROGRESS REPORT, SEPTEMBER 1, 1976--SEPTEMBER 1, 1978. Sadowski, G.S. (ed.). (Oak Ridge National Lab., TN (USA)). Nov 1978. Contract W-7405-ENG-26. 93p. AT.

Progress is summarized in the following LMFBR related studies: reactor surveillance and diagnostics; neutron sensor development; advanced fuel recycle program; and miscellaneous project design, development, and installation assistance.

- 2576 (ORNL--5490) NONDESTRUCTIVE TESTING DEVELOPMENT PROGRAM. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING SEPTEMBER 30, 1978. Hill, M.R. (Oak Ridge National Lab., TN (USA)). Dec 1978. Contract W-7405-ENG-26. 114p. AT.

Results of NDT development programs

sponsored by the Division of Reactor Research and Technology at Argonne National Laboratory, GE, Hanford Engineering Development Laboratory, and Oak Ridge National Laboratory are presented. Argonne's program consists of ultrasonic in-service inspection for the EBR-2 steam generators reliable examination techniques for coarse-grain SS especially welds, neutron laminagraphy tests on fuel assemblies, and ultrasonic wall thickness measurement in grooved tubes. GE reports on in-service inspection of the few tube test module and transition joints. Hanford describes radiographic eddy current, ultrasonic and hybrid methods and equipment. DRNL describes NDT for steam generators for the breeder reactor program, transition joints in pipes, and NDT development. (FS)

- 2577 (ORNL--5505) VITAMIN-E: AN ENDF/B-V MULTIGROUP CROSS-SECTION LIBRARY FOR LMFBR CORE AND SHIELD, LWR SHIELD, DOSIMETRY AND FUSION BLANKET TECHNOLOGY. Weisbin, C.R.; Roussin, R.W.; Wagschal, J.J.; White, J.E.; Wright, R.Q. (Oak Ridge National Lab., TN (USA)). Feb 1979. Contract W-7405-ENG-26. 27p. (ENDF--274). Dep. NTIS, PC A03/MF A01.

The Department of Energy (DOE) Office of Fusion Energy (OFE) and the Division of Reactor Research and Technology (DRRT) jointly sponsored the development of a coupled fine-group cross section library. This 171-neutron, 36-gamma-ray group library was based upon ENDF/B-IV and was intended to be applicable to fusion reactor neutronics and LMFBR core and shield analysis. Versions of the library are available from the Radiation Shielding Information Center (RSIC) at the Oak Ridge National Laboratory in both AMPX and CCC formats. Computer codes for energy group collapsing, interpolation on Bondarenko factors for resonance self-shielding and temperature corrections, and various other useful data manipulations are also available. The experience gained in the generation, validation and utilization of this library along with its broad range of applicability has led to the request for updating this data set using ENDF/B-V. Additional support in this regard has been provided by the Defense Nuclear Agency (DNA) and by the Electric Power Research Institute (EPRI) in support of weapons analyses and light water reactor shielding and dosimetry problems, respectively. The purpose of the report is to provide detailed specifications and rationale for the proposed ENDF/B-V update (designated VITAMIN-E) to the VITAMIN-C library.

- 2578 (ORNL--5512) INELASTIC RATCHETTING ANALYSIS OF THE 2 1/4 CR--1 MO STEEL TO TYPE 316 STAINLESS STEEL DISSIMILAR METAL WELDMENT REGION OF SPECIMEN TTT-3. Sartory, W.K. (Oak Ridge National Lab., TN (USA)). Mar 1979. Contract W-7405-ENG-26. 51p. AT.

The results of a ratchetting analysis of a dissimilar metal weldment between pipes of 2 1/4 Cr--1 Mo steel and type 316 stainless steel are presented and compared with earlier test results. The loading history consisted of 19 thermal downshocks at the inner surface, each followed by a sustained period under internal pressure at a temperature high enough for creep to occur in all three materials. Thermal analyses of the weldment region were performed, and the calculated outer-surface temperatures at the center plane of the weldment agree with the experiment. An inelastic structural analysis of the specimen was performed using constitutive equations. The agreement of the calculated strain with the experiment is about as close as was obtained earlier for the same pipe in regions far from any structural

discontinuities. Some limited creep damage results were also obtained.

operate in the creep range. (FS)

- 2579 (ORNL--5529) RECOMMENDATIONS CONCERNING MODELS AND PARAMETERS BEST SUITED TO BREEDER REACTOR ENVIRONMENTAL RADIOLOGICAL ASSESSMENTS. Miller, C.W.; Baes, C.F. III; Dunning, D.E. Jr. (Oak Ridge National Lab., TN (USA)). May 1980. Contract W-7405-ENG-26. 162p. NTIS, PC A08/MF A01.

Recommendations are presented concerning the models and parameters best suited for assessing the impact of radionuclide releases to the environment by breeder reactor facilities. These recommendations are based on the model and parameter evaluations performed during this project to date. Seven different areas are covered in separate sections.

- 2580 (ORNL--5538) INTERNAL BORE WELDING OF 2 1/4 CR-1 MO STEEL TUBE-TO-TUBESHEET JOINTS. Moorhead, A.J.; Reed, R.W. (Oak Ridge National Lab., TN (USA)). Jun 1979. Contract W-7405-ENG-26. 43p. AT.

We undertook a program to develop improved equipment for internal bore welding of the critical tube-to-tubesheet joints of steam generators and to gain understanding of the welding behavior of 2 1/4 Cr-1 Mo steel tubing by studying samples from four commercial heats of material. A commercial internal welding tip was improved by adding a Ta-10% W collet, which allowed use of a straight rather than bent tungsten electrode, and a fused quartz gas cup, which replaced three separate components. We found that the pressure differential (ΔP) between the inside and outside of the tubular weldments was critical in controlling weld bead geometry. With a slant tube manometer, we determined that ΔP held to 10 mm (0.4 in.) of water resulted in optimum bead contour. Several different procedures were evaluated for cleaning the tubing before welding. Weld porosity (determined by a very sensitive radiographic technique using a rod-anode microfocus x-ray source) was minimized by simple abrasion with a commercial mesh product followed by an alcohol rinse. Tubing from three of the four heats behaved very similarly in boreside welding; however, welds that were made in the fourth heat (which was from a VAR billet) under identical conditions as the others penetrated only about two-thirds of the tube wall. Consequently, we developed no single procedure that could be used to successfully weld all four materials. We conducted an extensive postweld heat treatment study on samples from a weld in each heat of tubing and measured the microhardness of various regions of the weldment in each sample. The heat-affected zones were harder than the fusion zones in these autogenous weldments. Furthermore, the weldments were harder than the base metals, even after lengthy thermal treatments at 677 or 732°C.

- 2581 (ORNL--5540) HIGH-TEMPERATURE STRUCTURAL DESIGN PROGRAM SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1978. Corum, J.M.; Pugh, C.E. (Oak Ridge National Lab., TN (USA)). Oct 1979. Contract W-7405-ENG-26. 320p. AT.

Progress is summarized in the ORNL High-Temperature Structural design program concerning constitutive equations for inelastic design analysis, inelastic structural tests and analyses, design criteria and rules for time-dependent deformation and failure, simplified design methods, thermal ratcheting tests, and structural design methods and weldments for LMFBR and other reactor system components that

- 2582 (ORNL--5548) FY 1977 PROGRESS REPORT OF THE CLINCH RIVER BREEDER REACTOR SOURCE RANGE FLUX MONITOR ASSESSMENT STUDY FOR THE REFERENCE HOMOGENEOUS CORE. Emmett, M.B.; Engle, W.W. Jr.; Flanagan, G.F.; Knee, H.E.; Loyd, D.L.; Selby, D.L.; Williams, L.R. (Oak Ridge National Lab., TN (USA)). Sep 1979. Contract W-7405-ENG-26. 38p. Dep. NTIS, PC A03/MF A01.

Interpretability of Inverse Kinetics Rod Drop (IKRD) was examined and found to be feasible. The absolute count rate at beginning-of-life (BOL) full shutdown was calculated and determined to be below minimum recommended levels. However, geometry and material changes in the SRFM were identified which could be employed to achieve the desired minimum levels. Generalized perturbation interpolation schemes to estimate detection efficiency changes were investigated and found to be feasible but not very practical. In addition, the uncertainties associated with the devised subcritical monitoring procedure were examined.

- 2583 (ORNL--5557) ANALYSIS OF THE CLINCH RIVER BREEDER REACTOR (CRBR) SOURCE RANGE FLUX MONITOR (SRFM) COUNT RATE HISTORY DURING REFUELING OF THE FIRST FOUR CYCLES. Selby, D.L.; Knee, H.E. (Oak Ridge National Lab., TN (USA)). Sep 1979. Contract W-7405-ENG-26. 45p. AT.

An annual shutdown for refueling is scheduled for all operating cycles of the Clinch River Breeder Reactor (CRBR). Operating procedures require that during refueling the Source Range Flux Monitor (SRFM) provide a warning to the operator to assure that the reactor does not approach criticality any closer than that level from which criticality could be attained by a single refueling error, with adequate margin for the associated uncertainties. The present procedure requires the use of a changing count rate alarm set point. This alarm set point would represent the subcriticality level corresponding to the worst single refueling error plus appropriate uncertainties. This report is a summary of the work performed at ORNL to determine and analyze the count rate history during each step of refueling through the first four cycles. In addition, relative alarm point values were determined and an effort was made to estimate the amount of conservatism associated with the method used to monitor refueling.

- 2584 (ORNL--5568) REVIEW OF METHODS FOR TREATMENT OF SOURCE EFFECTS IN THE MODIFIED SOURCE MULTIPLICATION METHOD FOR MONITORING THE REACTIVITY IN REFUELING THE CRBR. Mihalcz, J.T. (Oak Ridge National Lab., TN (USA)). Sep 1979. Contract W-7405-ENG-26. 13p. AT.

There are three methods by which source effects can be treated in the interpretation of modified source multiplication measurements in refueling a liquid-metal fast breeder reactor: (1) measurement of and correction for axial and radial source intensity distributions; (2) use of an additional source to override burnt-in source effects so they can be neglected, and (3) calculations. The implementation of these methods and their adequacy for the Clinch River Breeder Reactor (CRBR) is discussed. Of the three methods, the use of an additional source is the only method where an increase or decrease in the count rate definitely means a corresponding increase or decrease in reactivity.

2585 (ORNL--5579) METALS AND CERAMICS DIVISION ANNUAL PROGRESS REPORT, OCTOBER 1, 1978-JUNE 30, 1979. Peterson, S. (ed.). (Oak Ridge National Lab., TN (USA)). Sep 1979. Contract W-7405-ENG-26. 72p. Dep. NTIS, PC A04/MF A01.

Research is reported concerning: (1) engineering materials including materials compatibility, mechanical properties, nondestructive testing, pressure vessel technology, and welding and brazing; (2) fuels and processes consisting of ceramic technology, fuel cycle technology, fuels evaluation, fuels fabrication and metals processing; and (3) materials science which includes, ceramic studies, physical metallurgy and properties, radiation effects and microstructural analysis, metastable and superconducting materials, structure and properties of surfaces, theoretical research, and x-ray research and applications. Highlights of the work of the metallographic group and the current status of the High-Temperature Materials Laboratory (HTML) and the Materials and Structures Technology Management Center (MSTMC) are presented. (FS)

2586 (ORNL/BRP--78/3) MECHANICAL PROPERTIES TEST MATRICES ON TYPE 316 STAINLESS STEEL AND 2 1/4 CR--1 MO STEEL FOR U.S. DEPARTMENT OF ENERGY BREEDER REACTOR PROGRAM. Sikka, V.K. (comp.). (Oak Ridge National Lab., TN (USA)). Oct 1978. Contract W-7405-ENG-26. 99p. AT.

Details of the mechanical properties test program being undertaken by the United States Department of Energy on Structural Materials for Breeder Reactor Program are presented. The current program covers type 316 stainless steel and associated weld metal and 2 1/4 Cr--1 Mo steel and the transition joint weld materials and weldments. Section I of this report covers the tests to be made in air and after irradiation; Section II covers tests to be made after prior exposure in sodium and those to be made in static and dynamic sodium. The current (August 1978) status of various programs is presented.

2587 (ORNL/BRP--78/4) SPECIAL INFORMATION MEETING ON HIGH-TEMPERATURE STRUCTURAL DESIGN APPLICATIONS EXPERIENCE, GERMANTOWN, MARYLAND, SEPTEMBER 14--15, 1978. SUMMARY REPORT. (Oak Ridge National Lab., TN (USA)). Dec 1978. Contract W-7405-ENG-26. 514p. AT.

The special information meeting on high-temperature structural design applications experience was held September 14 to 15, 1978, at the Department of Energy in Germantown, Maryland. Although the presentations largely addressed liquid-metal-cooled breeder reactor applications, high-temperature gas-cooled reactor applications were addressed as well. This summary report was prepared to document the meeting. It contains, in addition to the Agenda and List of Attendees, a brief introduction to and summary of the meeting and a very brief overview and copy of the visuals for each of the presentations.

2588 (ORNL/BRP--79/3) NONDESTRUCTIVE TESTING DEVELOPMENT PROGRAM SEMIANNUAL PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1979. McClung, R.W. (comp.). (Oak Ridge National Lab., TN (USA)). Jul 1979. Contract W-7405-ENG-26. 94p. AT.

Nondestructive testing (NDT) research for LMFBR components is summarized. The work covered NDT techniques and equipment for inspection and monitoring of steam generators

and piping. The NDT techniques included ultrasonics, radiography, and eddy current testing.

2589 (ORNL/BRP--79/4) MODIFIED 9 CR-1 MO STEEL DEVELOPMENT PROGRAM PROGRESS REPORT, MARCH 31, 1979. Distefano, J.R. (comp.). (Oak Ridge National Lab., TN (USA)). Sep 1979. Contract W-7405-ENG-26. 210p. AT.

Research concerning Cr-Mo steel for use in LMFBR plant components is described. Creep, weld filler metal development, weldability, tensile testing, Charpy V-notch toughness studies, thermal aging, corrosion, hydrogen embrittlement, precipitate extraction and identification, and stress rupture testing are reported. (FS)

2590 (ORNL/BRP/CD--78/6) CLAD/DUCT MATERIALS DEVELOPMENT. TECHNICAL STATUS REPORT FOR JUNE 1978. Rowcliffe, A.F. (Oak Ridge National Lab., TN (USA)). Jun 1978. Contract W-7405-ENG-26. 3p. AT.

The objective of the program is to identify and develop advanced alloys for cladding and duct use with improved resistance to swelling and to irradiation-induced mechanical property degradation. Information on stainless steel-316 and V-series alloys is presented.

2591 (ORNL/BRP/M--78/6) BREEDER REACTOR PROGRAM: MATERIALS. TECHNICAL STATUS REPORT FOR JUNE 1978. Goodwin, G.M. (Oak Ridge National Lab., TN (USA)). Jun 1978. Contract W-7405-ENG-26. 21p. AT.

Breeder reactor materials are discussed. Data are presented concerning improved welds, steam generator materials, NDT, advanced breeder structural materials development, mechanical properties of structural materials for breeders, and fabrication of piping and pipe fittings. (FS)

2592 (ORNL/CSD--35) NEUTRONIC ANALYSIS OF LMFBRs DURING SEVERE CORE DISRUPTIVE ACCIDENTS. Tomlinson, E.T. (Oak Ridge National Lab., TN (USA)). Jan 1979. Contract W-7405-ENG-26. 48p. Dep. NTIS, PC A03/MF A01.

A number of numerical experiments were performed to assess the validity of diffusion theory and various perturbation methods for calculating the reactivity state of a severely disrupted liquid metal cooled fast breeder reactor (LMFBR). The disrupted configurations correspond, in general, to phases through which an LMFBR core could pass during a core disruptive accident (CDA). Two-reactor models were chosen for this study, the two zone, homogeneous Clinch River Breeder Reactor and the Large Heterogeneous Reactor Design Study Core. The various phases were chosen to approximate the CDA results predicted by the safety analysis code SAS3D. The calculational methods investigated in this study include the eigenvalue difference technique based on both discrete ordinate transport theory and diffusion theory, first-order perturbation theory, exact perturbation theory, and a new hybrid perturbation theory. Selected cases were analyzed using Monte Carlo methods. It was found that in all cases, diffusion theory and perturbation theory yielded results for the change in reactivity that significantly disagreed with both the discrete ordinate and Monte Carlo results. These differences were, in most cases, in a nonconservative direction.

2593 (ORNL/CSD/TM--35) CALCULATION OF THE REACTIVITY DUE TO BUBBLE COLLAPSE WITH THE

PROBABILITY TABLE METHOD. Hoffman, T.J.; Petrie, L.M. (Oak Ridge National Lab., TN (USA)). Dec 1977. Contract W-7405-ENG-26. 29p. Dep. NTIS, PC A03/MF A01.

The probability table method, originally developed for unresolved resonance calculations; is adapted to the calculation of bubble worths in a reactor core. Unlike previous approaches to this problem, this method is applicable to a broad class of problems, avoids a high variance Monte Carlo calculation, and preserves the main features of a random bubble distribution. This approach is illustrated by the solution to several problems of varying degrees of complexity.

2594 (ORNL/CSD/TM--38) TRANSPORT-DIFFUSION COMPARISONS FOR SMALL CORE LMFBR DISRUPTIVE ACCIDENTS. Tomlinson, E.T. (Oak Ridge National Lab., TN (USA)). Nov 1977. Contract W-7405-ENG-26. 51p. Dep. NTIS, PC A04/MF A01.

A number of numerical experiments were performed to assess the validity of diffusion theory for calculating the reactivity state of various small core LMFBR disrupted geometries. The disrupted configurations correspond, in general, to various configurations predicted by SAS3A for transient undercooling (TUC) and transient overpower (TOP) accidents for homogeneous cores and to the ZPPR-7 configurations for heterogeneous core. In all TUC cases diffusion theory was shown to be inadequate for the calculation of reactivity changes during core disassembly.

2595 (ORNL/MIT--274) RADIATIVE HEAT TRANSFER IN A 19-PIN SODIUM-VOIDED BUNDLE. Machnitz, M.; Budiman, B.; Roberts, Y.Y. (Massachusetts Inst. of Tech., Oak Ridge, TN (USA). School of Chemical Engineering Practice). 26 Apr 1978. Contract W-7405-ENG-26. 46p. Dep. NTIS, PC A03/MF A01.

A theoretical study was made of the radiative heat removal capability in a 19-pin bundle simulating the fuel assemblies of the commercial-scale (217-pin) Clinch River Breeder Reactor. Pin surface temperature distributions and the maximum allowable uniform pin power (without causing cladding meltdown) were determined for a range of emissivities and duct-wall temperatures. It was concluded that radiative heat transfer ranging to 0.1 kW/ft was comparable to free convection ranging to 0.02 kW/ft, and was lower than that of forced convection cooling to 2.17 kW/ft by the sodium vapor.

2596 (ORNL/MIT--285) ANALYSIS OF SBTF QUADELLIPTICAL FURNACES. Anderson, F.E.; Schulz, R.B. (Massachusetts Inst. of Tech., Oak Ridge, TN (USA). School of Chemical Engineering Practice). 23 Feb 1979. Contract W-7405-ENG-26. 43p. Dep. NTIS, PC A03/MF A01.

A computer model was developed which predicts the axial temperature profile and heat flux at the outer surface of the test section of the Sodium Boiling Test Facility constructed by the Engineering Technology Division at ORNL. The model was in agreement with observed temperature profiles at furnace power levels representative of single phase, dual phase, and dry-out operations. A parametric study demonstrated the effect of sodium flow rate and surface emissivities on the predicted temperature profile. It was concluded that axial conduction in the Hastelloy tube and sodium must be incorporated into the model to improve accuracy.

2597 (ORNL/NUREG/CR--0678) ANALYSIS OF THE HEAT AND MASS TRANSFER PROCESSES OF A UO₂ BUBBLE IN SODIUM FOR THE FUEL AEROSOL SIMULANT TEST (FAST). Tobias, M.L. (Oak Ridge National Lab., TN (USA)). 17 Apr 1979. Contract W-7405-ENG-26. 44p. (ORNL/NUREG/TM--307). Dep. NTIS, PC A03/MF A01.

The anticipated behavior of uranium oxide vapor bubbles produced by the capacitor discharge vaporization (CDV) method in the Fuel Aerosol Simulant Test (FAST) Facility is discussed on the basis of relatively simple physical models. Results of calculations for the rate of bubble rise and for heat and mass transfer rates are presented. Parametric studies indicate that future analysis efforts should emphasize the diffusion condensation process and the loss of heat from the bubble by radiation. Transfer of heat in the surrounding sodium is rapid enough that simplified models should be adequate. No important effects were noted in connection with bubble depth, initial quantity of UO₂, or initial superheat.

2598 (ORNL/NUREG/NSIC--139) LMFBR SAFETY. 5. REVIEW OF CURRENT ISSUES AND BIBLIOGRAPHY OF LITERATURE (1975--1976). Buchanan, J.R.; Kailholtz, G.W. (Oak Ridge National Lab., TN (USA)). 8 Jun 1977. Contract W-7405-ENG-26. 575p. (ORNL/NUREG/NSIC--125(Vol.5)). Dep. NTIS, PC A25/MF A01.

The current status of liquid-metal fast breeder reactor (LMFBR) development and one of the principal safety issues, a hypothetical core-disruptive accident (HCDA), are discussed. Bibliographic information on worldwide LMFBRs relative to the development and safety of the breeder reactor is presented for the period 1975 through 1976. The bibliography consists of approximately 1618 abstracts covering early research and development and operating experiences leading up to the present design practices that are necessary for the licensing of breeder reactors. Keyword, author, and permuted-title indexes are included for completeness.

2599 (ORNL/NUREG/TM--124) SOURCE TERM ASSESSMENT CONCEPTS FOR LMFBRs: AEROSOL RELEASE AND TRANSPORT (ART) ANALYTICAL PROGRAM. Kress, T.S. (Oak Ridge National Lab., TN (USA)). Nov 1977. Contract W-7405-ENG-26. 72p. Dep. NTIS, PC A04/MF A01.

Analytical needs for LMFBR HCDA source term assessment are outlined and the physical events connected with the formation of an aerosol source are discussed. The important features of the movement of hazardous materials from the core site to the boundaries of the secondary containment are described, with accompanying brief reviews of prior work. The overall problem is broken down into tractable parts and approaches for attacks on the various problems are suggested. Topics include the state of the disrupted core, interactions with intact above-core structure, HCDA bubble formation, heat exchange with the sodium pool, aerosol formation by nucleation and fragmentation, attenuation effects, and aerosol behavior in secondary containments.

2600 (ORNL/NUREG/TM--129) FUEL AEROSOL SIMULANT TEST (FAST) PLAN. Wright, A.L.; Smith, A.M.; Kress, T.S. (Oak Ridge National Lab., TN (USA)). 14 Sep 1977. Contract W-7405-ENG-26. vp. Dep. NTIS, PC A02/MF A01.

This report presents the work plan for the Fuel Aerosol Simulant Test (FAST), which constitutes task 5 of the Aerosol Release and Transport (ART) Program, sponsored by the

Division of Reactor Safety Research of the Nuclear Regulatory Commission. These tests involve the study of the formation, rise, and condensation of a surrogate LMFBR fuel (UO_2) vapor bubble produced in liquid sodium to simulate phenomena associated with hypothetical core-disruptive accidents (HCDAs). UO_2 bubbles will be produced by the capacitor discharge vaporization (CDV) technique. Tests through FY 1979 will be done in argon, water, and in sodium; results will be analyzed using models developed in the ART program.

- 2601 (ORNL/NUREG/TM--142) BREEDER REACTOR AEROSOL RELEASE AND TRANSPORT PROGRAM. QUARTERLY PROGRESS REPORT, APRIL--JUNE 1977. Fontana, M.H.; Kress, T.S. (Oak Ridge National Lab., TN (USA)). 16 Nov 1977. Contract W-7405-ENG-26. 65p. Dep. NTIS, PC A04/MF A01.

The report summarizes studies designed to investigate radionuclide release and transport from breeder reactors for reactor events of severity up to and including hypothetical core-disruptive accidents (HCDAs). Experimental program topics discussed include (1) CDV-CRI-III design considerations and experimental operation; (2) fuel aerosol characterization studies in the CRI-II facility; (3) NSPP sodium oxide aerosol experiments; (4) FAST facility design, site preparation, and component procurement and fabrication; and (5) exploratory CDV experiments in support of the FAST program. Analytical program topics discussed include an idealized expansion calculational model and a model for UO_2 vapor condensation on cooler structural surfaces.

- 2602 (ORNL/NUREG/TM--148) MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT OAK RIDGE NATIONAL LABORATORY, AUGUST 1977. Fee, G.G. (comp.). (Oak Ridge National Lab., TN (USA)). 13 Sep 1977. Contract W-7405-ENG-26. 30p. Dep. NTIS, PC A03/MF A01.

Technical highlights are presented for the following safety-related studies: heavy section steel technology, fission product beta and gamma energy release, fission product release from LWR fuel, fission product transport tests, multirod burst tests, Nuclear Safety Information Center, PWR blowdown heat transfer-separate effects, zircaloy fuel cladding collapse studies, zirconium metal-water oxidation kinetics, aerosol release and transport from LMFBR fuel, HTGR safety analysis and research, design criteria for piping and nozzles, and noise diagnostics for safety assessment.

- 2603 (ORNL/NUREG/TM--158) MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT OAK RIDGE NATIONAL LABORATORY. Fee, G.G. (comp.). (Oak Ridge National Lab., TN (USA)). 11 Oct 1977. Contract W-7405-ENG-26. 25p. Dep. NTIS, PC A03/MF A01.

Brief highlights of technical progress during September 1977 are presented for thirteen separate program activities which comprise the ORNL research program for the Office of Nuclear Regulatory Research's Division of Reactor Safety Research.

- 2604 (ORNL/NUREG/TM--160) DEVELOPMENT OF THE CAPACITOR DISCHARGE VAPORIZATION TECHNIQUE TO PRODUCE AEROSOLS FORMED UNDER CONDITIONS POSTULATED FOR HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS. Kelly, M.J.; Parker, G.W.; Rochelle, J.M. (Oak Ridge National Lab., TN

(USA)). 22 Jun 1978. Contract W-7405-ENG-26. 94p. (NUREG/CR--0226). Dep. NTIS, PC A05/MF A01.

This report describes the development of a Capacitor Discharge Vaporization (CDV) method for the out-of-pile creation of energy states in fast breeder reactor fuel (or simulants) that are consistent with those postulated for Hypothetical Core Disruptive Accidents (HCDAs). The results of the development experiments, concurrent planning, test article construction, peripheral equipment design and construction, and computer modeling studies are discussed in relation to their effects on the design of the Oak Ridge National Laboratory's Capacitor Discharge Vaporization (ORNL CDV) System. The completed ORNL CDV System is described in detail, including all the required subsystems and data acquisition equipment. Operational features, equipment capability, and expected performance are also discussed.

- 2605 (ORNL/NUREG/TM--165) INVENTORY AND RELATIVE BIOLOGICAL HAZARD OF LMFBR CORE MATERIALS. Tobias, M.L. (Oak Ridge National Lab., TN (USA)). 16 Jan 1978. Contract W-7405-ENG-26. 15p. Dep. NTIS, PC A02/MF A01.

Information on the composition of an LMFBR core by element has been collected on the basis of published design data and ORIGEN calculations. The mass quantities of structural materials, actinides, and fission products are in the ratio of 1:1.4:0.1 with actinides, particularly plutonium and neptunium, representing the major inhalation and ingestion hazard. Some of the fission products are present in quantities large enough to suggest that they warrant specific attention in source term analysis.

- 2606 (ORNL/NUREG/TM--172) NOVEMBER 1977 MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT OAK RIDGE NATIONAL LABORATORY. Mynatt, F.R. (comp.). (Oak Ridge National Lab., TN (USA)). 12 Dec 1977. Contract W-7405-ENG-26. 31p. Dep. NTIS, PC A04/MF A01.

Brief highlights are presented for the following safety-related programs: heavy section steel technology, fission product beta and gamma energy release, fission product release from LWR fuel, fission product transport tests, multirod burst tests, Nuclear Safety Information Center, PWR blowdown heat transfer-separate effects, Zircaloy fuel cladding collapse studies, aerosol release and transport from LMFBR fuel, advanced instrumentation for reflood studies, HTGR safety analysis and research, design criteria for piping and nozzles, and noise diagnostics for safety assessment.

- 2607 (ORNL/NUREG/TM--173) BREEDER REACTOR AEROSOL RELEASE AND TRANSPORT PROGRAM QUARTERLY PROGRESS REPORT FOR JULY--SEPTEMBER 1977. Kress, T.S. (Oak Ridge National Lab., TN (USA)). Apr 1978. Contract W-7405-ENG-26. 54p. Dep. NTIS, PC A04/MF A01.

The report summarizes progress for the Aerosol Release and Transport Program. The program is designed to investigate radionuclide release and transport from breeder reactors for reactor events of severity up to and including hypothetical core-disruptive accidents (HCDAs). Experimental program topics discussed include fuel aerosol characterization studies in the CRI-II facility; NSPP sodium oxide pool fire and spray fire aerosol experiments; FAST facility design, component procurement and fabrication, site preparation, and CRI-III facility site change and reinstallation.

Analytical program topics discussed include a bubble condensation model for FAST experiments and discussion of breeder reactor core contents which can contribute to an aerosol generated by a hypothetical core-disruptive accident (HCDA).

- 2608 (ORNL/NUREG/TM--179) SODIUM OXIDE AEROSOL STUDY: NSPP RUNS 101 TO 105, DATA RECORD REPORT. Adams, R.E.; Kress, T.S.; Parsly, L.F. Jr. (Oak Ridge National Lab., TN (USA)). 9 Mar 1978. Contract W-7405-ENG-26. 83p. Dep. NTIS, PC A03/MF A01.

This data record report summarizes the first five sodium oxide aerosol tests conducted in the Nuclear Safety Pilot Plant (NSPP) project at Oak Ridge National Laboratory. The goal of the project is to establish the validity (or level of conservatism) of the aerosol behavioral code, HAARM-2, and follow-on codes under development at the Battelle Columbus Laboratories for the U.S. Nuclear Regulatory Commission (NRC). Included in the report are descriptions of the five tests together with graphs and tables summarizing the results.

- 2609 (ORNL/NUREG/TM--182) DECEMBER 1977 MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT OAK RIDGE NATIONAL LABORATORY. Mynatt, F.R. (comp.). (Oak Ridge National Lab., TN (USA)). 12 Jan 1978. Contract W-7405-ENG-26. 28p. Dep. NTIS, PC A03/MF A01.

Technical highlights are presented for the following safety-related studies: heavy section steel technology, fission product release from LWR fuel, fission product transport tests, multirod burst tests, Nuclear Safety Information Center, PWR blowdown heat transfer-separate effects, Zircaloy fuel cladding collapse studies, aerosol release and transport from LMFBR fuel, advanced instrumentation for reflood studies (AIRS), HTGR safety analysis and research, design criteria for piping and nozzles, and noise diagnostics for safety assessment. (DG)

- 2610 (ORNL/NUREG/TM--193) BREEDER REACTOR AEROSOL RELEASE AND TRANSPORT PROGRAM. QUARTERLY PROGRESS REPORT FOR OCTOBER--DECEMBER 1977. Kress, T.S. (Oak Ridge National Lab., TN (USA)). 25 Apr 1978. Contract W-7405-ENG-48. 36p. (NUREG/CR--0090). Dep. NTIS, PC A03/MF A01.

This report summarizes progress for the Aerosol Release and Transport Program, sponsored by the Division of Reactor Safety Research of the Nuclear Regulatory Commission, for the period October--December 1977. The program is designed to investigate radionuclide release and transport from breeder reactors for reactor events of severity up to and including hypothetical core-disruptive accidents (HCDAs). Experimental program topics discussed include (1) fuel aerosol characterization studies in the CRI-II facility; (2) design and development of a mixed sodium oxide-uranium oxide generating system; (3) the first uranium oxide experiment in the Nuclear Safety Pilot Plant; (4) Fuel Aerosol Simulation Test Facility installation progress; and (5) progress in the relocation of the CRI-III facility. From the analytical program, calculations of the fuel vapor quality and bubble diameter resulting from expansion of saturated liquid fuel to a lower pressure and simultaneous evaporation of varying amounts of sodium are reported.

- 2611 (ORNL/NUREG/TM--197) FEBRUARY 1978 MONTHLY HIGHLIGHTS FOR OFFICE OF NUCLEAR

REGULATORY RESEARCH PROGRAMS AT OAK RIDGE NATIONAL LABORATORY. Mynatt, F.R. (comp.). (Oak Ridge National Lab., TN (USA)). 10 Mar 1978. Contract W-7405-ENG-26. 37p. Dep. NTIS, PC A03/MF A01.

Brief highlights are presented for the following safety-related studies: heavy section steel technology, fission product beta and gamma energy release, fission product release from LWR fuel, multirod burst tests, Nuclear Safety Information Center, PWR blowdown heat transfer--separate effects, Zircaloy fuel cladding collapse studies, aerosol release and transport from LMFBR fuel, advanced instrumentation for reflood studies, HTGR safety analysis and research, design criteria for piping and nozzles, noise diagnostics for safety assessment, improved eddy current inservice inspection for steam generator tubing, LWR pressure vessel irradiation, and NRC measured data repository.

- 2612 (ORNL/NUREG/TM--201) MARCH 1978 MONTHLY HIGHLIGHTS, OFFICE OF NUCLEAR REGULATORY RESEARCH PROGRAMS AT OAK RIDGE NATIONAL LABORATORY. Myatt, F.R. (comp.). (Oak Ridge National Lab., TN (USA)). Apr 1978. Contract W-7405-ENG-26. 40p. Dep. NTIS, PC A03/MF A01.

Technical highlights are presented for the following safety-related studies: heavy section steel technology, fission product beta and gamma energy release, fission product release from LWR fuel, fission product transport tests, multirod burst tests, nuclear safety information center, PWR blowdown heat transfer-separate effects, Zircaloy fuel cladding collapse studies, aerosol release and transport from LMFBR fuel, advanced instrumentation for reflood studies, HTGR safety analysis and research, design criteria for piping and nozzles, noise diagnostics for safety assessment, improved eddy current in-service inspection for steam generator tubing, light water reactor pressure vessel irradiation program, and the NRC measured data repository. (DG)

- 2613 (ORNL/RSIC--41, pp 107-119) EXPERIENCE IN DEVELOPING AND USING THE VITAMIN-C 171-NEUTRON, 36-GAMMA-RAY GROUP CROSS-SECTION LIBRARY. Roussin, R.W.; Weisbin, C.R. White, J.E.; Wright, R.Q.; Greene, N.M.; Ford, W.E. III; Wright, J.B.; Diggs, B.R. Oct 1978.

From Radiation shielding information center seminar; Oak Ridge, TN, USA (14 Mar 1978).

The Department of Energy (DOE) Division of Magnetic Fusion Energy (DMFE) and Reactor Research and Technology (DRRT) jointly sponsored the development of a coupled, fine-group cross-section library. The 171-neutron, 36-gamma-ray group library is intended to be applicable to fusion reactor neutronics and LMFBR core and shield analysis. Versions of the library are available from the Radiation Shielding Information Center (RSIC) at Oak Ridge National Laboratory in both AMPX and CCCC formats. Computer codes for energy group collapsing, interpolation on Bondarenko factors for resonance self-shielding and temperature corrections, and various other useful data manipulations are available. The experience gained in the utilization of this library is discussed. Indications are that this venture, which is designed to allow users to derive problem-dependent cross sections from a fine-group master library, has been a success.

- 2614 (ORNL/RSIC--42, pp 251-279) SENSITIVITY THEORY FOR REACTOR THERMAL-HYDRAULICS PROBLEMS. Oblov, E.M. (Oak Ridge

National Lab., Oak Ridge, TN). Feb 1979.

From Workshop on theory and application of sensitivity and uncertainty analysis; Oak Ridge, TN, USA (22 Aug 1978).

A sensitivity theory based on reactor physics experience was successfully developed for a reactor thermal-hydraulics problem. The new theory is derived for the case of non-linear, transient heat and mass transfer in a typical reactor subassembly. Suitable adjoint equations for heat and fluid flow are presented along with methods for deriving the sources and boundary and final conditions for these equations. Expressions for the sensitivity of any integral temperature response to problem input data are also presented. The theory is applied to a sample problem describing the steady-state thermal-hydraulic conditions in a CRBR fuel channel. For this case, sensitivity coefficients are derived for several thermal response functions (i.e. peak clad and peak fuel temperature) for all physical input data (i.e. the heat transfer coefficient, thermal conductivities, etc.). A typical uncertainty analysis for peak clad and peak fuel temperature was also performed using uncertainty information about the physical data. Conclusions are drawn about the applicability of this approach to more general problems and the procedures for its implementation in conjunction with large safety or thermal-hydraulics codes are outlined. The method is also compared with currently used response surface techniques.

- 2615 (ORNL/SUB--79/55455/1) IMPLEMENTATION OF THE UNIFIED PLASTICITY--CREEP CONSTITUTIVE EQUATIONS WITH THE FINITE ELEMENT METHOD. Stoneking, J.E.; Baker, A.J.; Levine, D.L.; Soliman, M.O. (Tennessee Univ., Knoxville (USA), Dept. of Engineering Science and Mechanics). 12 Apr 1979. Contract W-7405-ENG-26. 62p. AT.

The implementation of unified plasticity--creep constitutive models into production finite element structural analysis codes requires identification of efficient numerical integration procedures for solution of the resultant, highly nonlinear ordinary differential equation systems. A standard implicit integration algorithm has been embedded within a Newton iteration procedure and evaluated for solution of the unified model applied to an axisymmetric example problem. A computer code was written using the elementary constant-strain triangular finite element. Numerical results indicate the developed procedure is fully capable of solving the sample problem with reasonable numerical efficiency.

- 2616 (ORNL/Sub--4308/3) STATUS OF INCOLOY ALLOY 800 DEVELOPMENT FOR BREEDER REACTOR STEAM GENERATORS. DiStefano, J.R. (comp.). (Westinghouse Electric Corp., Tampa, FL (USA), Tampa Div.). Dec 1978. Contract W-7405-ENG-26. 278p. AT.

Development of Incoloy alloy 800 for use in breeder reactor steam generators has been discontinued. Previously unpublished work on the program is presented as a collection of three reports: Study of the Softening and Recrystallization Response of Cold Worked and Annealed Alloy 800, ASME and B and PV Code Recommendations of Design Stresses for Use of Annealed Alloy 800 in Elevated Temperature Nuclear Vessels, and Relation of Physical Metallurgy to Mechanical Creep Behavior of Alloy 800.

APPLICATION OF THE PLANS COMPUTER PROGRAM FOR ANALYSIS OF NUCLEAR REACTOR STRUCTURAL COMPONENTS. Levy, A.; Pifko, A.; Armen, H. Jr. (Grumman Aerospace Corp., Bethpage, NY (USA), Research Dept.). Jun 1977. Contract W-7405-ENG-26-4485. 65p. (RE--542). AT.

PLANS is a group of finite element computer programs for the nonlinear analysis of complex structures. Four modules of the PLANS group described in the report are BEND, for the elastic-plastic analysis of built-up structures where bending and membrane effects are significant; HEX, for the three-dimensional elastic-plastic analysis of general solids; REVBY, for the elastic-plastic analysis of axisymmetrically loaded bodies of revolution; and OUT-OF-PLANE-GM, for the elastic-plastic and geometric nonlinear analysis of built-up sheet-stringer type structures. The capabilities and present status of these modules are summarized. All of the modules in PLANS employ the 'initial strain' concept within an incremental procedure to account for the effect of plasticity and include the capability for cyclic analysis. Geometric nonlinearities are included in OUT-OF-PLANE-GM using the 'updated' or convected coordinate approach.

- 2618 (ORNL/SUB--4485/2) DEVELOPMENT OF THE PLANS COMPUTER PROGRAM FOR ELASTIC--PLASTIC--CREEP ANALYSIS OF NUCLEAR REACTOR STRUCTURAL COMPONENTS. Levy, A. (Grumman Aerospace Corp., Bethpage, NY (USA), Research Dept.). Jan 1979. Contract W-7405-ENG-26. 34p. AT.

This report summarizes the continued development of computationally efficient finite-element methods for accurately predicting the isothermal three-dimensional elastic--plastic--creep responses of thick and thin shell structures subjected to mechanical and thermal loads. This work is directed toward verifying the structural integrity of liquid metal fast breeder reactor (LMFBR) components in the creep regime. The most recent developments include the use of multistress points within an element, temperature variation within an element, combined time dependent creep and isothermal plasticity, and restart capabilities.

- 2619 (ORNL/Sub--7322/1) MORE EFFICIENT CREEP RATCHETING BOUNDS. Porowski, J.S.; O'Donnell, W.J. (O'Donnell and Associates, Inc., Pittsburgh, PA (USA)). Oct 1978. Contract W-7405-ENG-26. 46p. AT.

The O'Donnell-Porowski upper bound solutions for creep ratcheting are extended to include material-hardening effects and temperature-dependent yield properties. Energy methods are introduced in order to obtain more efficient bounds for loading histograms involving a limited number of severe cycles in the plastic ratcheting regime, interspersed with cycling of lesser magnitude.

- 2620 (ORNL/SUB--7463/1) SIMPLIFIED INELASTIC ANALYSES USING ONE-DIMENSIONAL MODELS. Kamal, S.A.; Chern, J.M. (Foster Wheeler Energy Corp., Livingston, NJ (USA)). Sep 1979. Contract W-7405-ENG-26. 106p. AT.

A method of inelastic analysis using simplified structural models is investigated with special emphasis on the predictions of creep-fatigue damage and inelastic strain accumulation. The simplified models considered include the generalized plane strain, axisymmetric thick-walled cylinder which is applicable when the thermal loading in the

- 2617 (ORNL/Sub--4485/1) DEVELOPMENT AND

actual structure is dominated by the through-the-wall temperature gradient, and the finite cylindrical shell which is appropriate when the meridional temperature gradient becomes dominant. From the thermal elastic analysis results of the detailed finite element model, the critical areas are identified, the dimensions of the simplified model selected, and the associated loading histograms derived. Inelastic analyses using efficient computer codes are then performed to simulate the stress-strain conditions of these critical areas. The method is applied to the primary inlet nozzle and Y-junction of the CRBRP-IHX for which detailed inelastic analysis results for unit load cycles are available for comparison. It is indicated in the first application that the generalized plane strain model is able to yield accurate and conservative predictions. In the second application in which the cylindrical shell model is employed with specified displacement boundary conditions, the general trend of stress variation in a complete load cycle can be simulated and the fatigue predictions were conservative. However, it was found that the rapid stress relaxation tends to underestimate the creep damage and inelastic strain accumulations. The end-of-life creep damage prediction becomes conservative if the trend of cyclic change is taken into consideration; the inelastic strain prediction is persistently non-conservative by a range of 27 to 81%. A procedure to improve predictions that combines the force and displacement boundary conditions is outlined.

- 2621 (ORNL/Sub--7644/10) RESEARCH IN REMOTE SIGNAL TRANSMISSION ELECTRONICS. Kennedy, E.J.; Blalock, T.V.; Phelps, J.; Rochelle, R.; Dyer, F. (Tennessee Univ., Knoxville (USA)). Dept. of Electrical Engineering. 1 Oct 1979. Contract W-7405-ENG-26. 89p. (TR-EE/EL--10). Dep. NTIS, PC A05/MF A01.

Research has been directed toward the evaluation of semiconductor circuits suitable for operation at temperatures of 250°C to 300°C. These circuits are to be utilized in monitoring neutron flux in a breeder reactor. Worst-case performance evaluations are made on an amplifier-discriminator circuit utilizing commercially available operational amplifier chips incorporated in a hybrid thick-film package, which is suitable for use to 250°C. Another hybrid thick-film amplifier-discriminator system was developed and tested for use to 300°C. A dc-dc converter to supply +10v and -10v for amplifier power was designed, and evaluated up to an operating temperature of 300°C. Detailed design theory equations for high temperature dc-dc converters are presented. Significant problems with ceramic capacitors utilized at 300°C were found.

- 2622 (ORNL-TM--3689) LMFBR SPENT FUEL TRANSPORT: CONCEPTUAL DESIGN AND PARTIAL SAFETY ANALYSIS OF A SODIUM-COOLED CASK. Irvine, A.R.; Shappert, L.B.; Evans, J.H.; Canonico, D.A. (Oak Ridge National Lab., TN (USA)). Feb 1972. Contract W-7405-ENG-26. 80p. Dep. NTIS, PC A05/MF A01.

Conceptual designs for 6- and 18-subassembly casks are presented. The casks are intended for transport of LMFBR spent fuel which has decayed a minimum of 30 days. These casks use sodium as the primary coolant, an auxiliary shield coolant system in normal operation, heavy steel members as both gamma shield and structure, and a eutectic mixture of LiOH and NaOH as a neutron shield. The analysis indicates that there will be no leakage of coolant or fission products under normal or hypothetical accident conditions.

- 2623 (ORNL-TM--4283) ORNL TSF PIPE CHASE NEUTRON STREAMING EXPERIMENT: PHASE TWO. McGregor, B.J.; Clifford, C.E.; Muckenthaler, F.J.; Mymatt, F.R. (Oak Ridge National Lab., TN (USA)). Aug 1975. Contract W-7405-ENG-26. 51p. Dep. NTIS, PC A04/MF A01.

A second ORNL Tower Shielding Facility experiment has been performed and analyzed in the investigation of techniques for predicting the extent of neutron streaming through the sodium coolant pipes of the Fast Flux Test Facility (FFTF). Measurements of neutron intensities were made with Bonner ball detectors, and neutron spectra were measured with a Benjamin spectrometer. The analysis consisted of calculations of the Bonner ball and spectrometer responses with the two-dimensional discrete ordinates code DOT-III using 50-group and 14-group cross-section sets and assumptions of both isotropic and anisotropic boundary sources at the pipe-chase entrance. The Monte Carlo codes AMC and MORSE were used to correct the DOT calculations for three-dimensional geometric effects inherent in the experiment. For points inside the plenum, the calculated and experimental Bonner ball responses are in exceptionally good agreement; for points outside the plenum the calculated responses are approximately 20% higher than the experimental results. The calculated and measured neutron spectra agree at all three locations. It is concluded that the same calculational approach would accurately predict, or possibly slightly overpredict, the neutron flux levels that will exist in the heat exchanger vaults of the FFTF.

- 2624 (ORNL/TM--5836) EVALUATION OF INITIAL FUEL LOADING MONITORING IN THE CLINCH RIVER BREEDER REACTOR PLANT. Ackermann, N.J. Jr.; Booth, R.S. (Oak Ridge National Lab., TN (USA)). Dec 1977. Contract W-7405-ENG-26. 14p. AT.

Methods to monitor the initial core loading of the CRBRP are evaluated and ranked in terms of their sensitivity, interpretability, ease of implementation, and cost. This monitoring is an essential element of the administrative control of the initial fuel loading operation, which is the basic protection against an inadvertent criticality and its attendant safety problems. The recommended method for monitoring the initial fuel loading operation involves the addition of a special neutron source to the central region of the core during the initial fuel loading and the use of ex-vessel neutron detectors (referred to as 'source enhanced, ex-vessel detector method'). This method resolves the refueling interference problem of the in-core detector method and the poor sensitivity problem of the ex-vessel detector method.

- 2625 (ORNL/TM--5865/V5) EXPERIMENTAL ENGINEERING SECTION SEMI-ANNUAL PROGRESS REPORT, MARCH 1, 1976--AUGUST 31, 1976. VOLUME 5. REACTOR PROGRAMS. Ryon, A.D. (comp.). (Oak Ridge National Lab., TN (USA)). Mar 1978. Contract W-7405-ENG-26. 168p. AT.

The work covered LMFBR Iodex tests, voloxidizer studies (rotary kiln residence-time distribution and heat transfer tests), HTGR fuel refabrication, HTGR off-gas processing, and leaching of radium with nitric acid from uranium ores and tailings. (DLC)

- 2626 (ORNL/TM--5868) CREEP-PLAST II USER'S MANUAL. Clinard, J.A. (Oak Ridge National Lab., TN (USA)). May 1978. Contract W-7405-ENG-36. 122p. AT.

This report summarizes program operations and provides necessary user information to implement CREEP-PLAST II, a finite-element computer program for elastic-plastic-creep analysis of two-dimensional (plane stress or axisymmetric) structures.

- 2627 (ORNL/TM--5889) STATE OF THE ART OF FUEL PIN SIMULATORS FOR LMFBR OUT-OF-REACTOR SAFETY TESTS; STATUS REPORT. Baucum, W.E.; MacPherson, R.E.; Clark, D.L.; McCulloch, R.W. (Oak Ridge National Lab., TN (USA)). 22 Aug 1977. Contract W-7405-ENG-26. 178p. AT.

The liquid-metal fast breeder reactor (LMFBR) program out-of-reactor tests require high-performance, electrically heated fuel pin simulators (FPSs) to duplicate the configuration, heat flux profiles, and operational capabilities of nuclear fuel pins. Fuel pin simulators having a maximum heat flux of 315 W/cm^2 ($10^6 \text{ Btu hr}^{-1} \text{ ft}^{-2}$) in 1000°C (1830°F) sodium (category II) are considered state of the art when fabricated using conventional BN powder filling techniques, provided appropriate materials and fabrication techniques are used. In addition, fuel pin simulators of the same geometry for continuous operation in boiling sodium at 1000°C and having the capability for surviving for a short interval after void initiation and film dryout (category III) are also required in many test procedures. The results presented here have provided guidance for procurement, inspection, and assembly of fuel pin simulators for eight 19-pin bundles which have been extensively tested in the THORS facility.

- 2628 (ORNL/TM--5906) HEAT-TO-HEAT VARIATION OF TENSILE PROPERTIES OF ANNEALED 2 1/4 CR-1 MO STEEL. Klueh, R.L. (Oak Ridge National Lab., TN (USA)). Jul 1977. Contract W-7405-ENG-26. 40p. AT.

The tensile properties of twelve heats of 2 1/4 Cr-1 Mo steel were examined over the range 25 to 566°C . The twelve heats included: five air-melted tubing heats, two pieces from vacuum-arc remelted (VAR) forgings, two pieces from electroslag remelted (ESR) plates, two air-melted plates, and one air-melted forging. Tests were made on the steel in the as-received condition (vendor anneal) and, in order that all heats could be compared from a common basis, material from each of the heats was tested after it was annealed in our laboratory. For the twelve heats of steel tested, large property variations were observed. Decided differences were noted between the air-melted tubing, the air-melted plates and forging, and the ESR and VAR steels. Explanations for these differences are offered in terms of the types of precipitation reactions that occur in these steels at elevated temperatures.

- 2629 (ORNL/TM--5940) LMFBR SAFETY AND CORE SYSTEMS PROGRAMS. PROGRESS REPORT, JANUARY--MARCH 1977. Fontana, M.H.; Wantland, J.L. (Oak Ridge National Lab., TN (USA)). 25 Jul 1977. Contract W-7405-ENG-26. 94p. AT.

Topics discussed are program coordination; THORS analysis and data evaluation; detection methods development; THORS facility operation; materials for fuel pin simulation; the sodium boiling test facility; neutronics analysis of a disrupted core; central computerized data base for LMFBR safety codes; model evaluation of LMFBR reactivity releases; LMFBR reliability data analysis center; and environmental assessment of advanced LMFBR fuels.

PLASTICITY MODEL FOR STRUCTURAL METALS AT HIGH TEMPERATURE. Robinson, D.N. (Oak Ridge National Lab., TN (USA)). Oct 1978. Contract W-7405-ENG-26. 62p. AT.

A constitutive theory for use in high-temperature structural analysis of breeder reactor components is presented. It is unified in the sense that plastic and creep strains are considered to arise from similar physical mechanisms. The theory is presented in multiaxial terms along with a discussion of its underlying assumptions, its relationship to other theories and the experimental observations upon which it is based and is intended to model. The results of an extensive parameter sensitivity study are reported which provide understanding of the influence of the various model parameters on the critical features of predicted response.

- 2631 (ORNL/TM--5976) NUCLEAR SHIPPING AND WASTE DISPOSAL COST ESTIMATES. Hudson, C.R. II. (Oak Ridge National Lab., TN (USA)). Nov 1977. Contract W-7405-ENG-26. 24p. Dep. NTIS, PC A03/MF A01.

Cost estimates for the shipping of spent fuel from the reactor, shipping of waste from the reprocessing plant, and disposal of reprocessing plant wastes have been made for five reactor types. The reactors considered are the light-water reactor (LWR), the mixed-oxide-fueled light-water reactor (MOX), the Canadian deuterium-uranium reactor (CANDU), the fast breeder reactor (FBR), and the high-temperature gas-cooled reactor (HTGR). In addition to the cost estimates, this report provides details on the bases and assumptions used to develop the cost estimates.

- 2632 (ORNL/TM--5992) ANALYSES OF TSF EXPERIMENTS ON RADIATION HEATING IN A STAINLESS STEEL-SODIUM CRBR RADIAL SHIELD MOCKUP USING A 32-INCH DIAMETER COLLIMATED BEAM SOURCE. Maerker, R.E. (Oak Ridge National Lab., TN (USA)). Oct 1977. Contract W-7405-ENG-26. 60p. AT.

Analyses of the second phase of the radiation heating experiments performed at the Tower Shielding Facility have been made where the emphasis was placed on investigating the accuracy of TLD measurements in mockups of the CRBR radial shield. In particular, it is concluded that with the aid of flux calculations described in this report the corrections to both the LiF and CaF_2 TLD measurements are well enough understood that the TLD measurements can be brought into $\pm 3\%$ agreement with both ion chamber measurements and with each other. The accuracy of the gamma-ray calculations is limited by the accuracy of the 51 neutron group transport calculations; the production cross sections present in ENDF/B-IV are adequate for ^{235}U , sodium, and stainless steel.

- 2633 (ORNL/TM--5993) ADVANCED FUEL RECYCLE PROGRAM PROGRESS REPORT, APRIL 1--JUNE 30, 1977. Unger, W.E. (comp.). (Oak Ridge National Lab., TN (USA)). Sep 1977. Contract W-7405-ENG-26. vp. AT.

This quarterly progress report on the Fuel Recycle Program covers the ongoing development program aimed at providing the technology for production-scale advanced fuels recycle. The continuing basic laboratory and hot-cell studies, engineering research, and component equipment development activities are in support of a cold integrated equipment test scheduled for 1981 and a current conceptual design of a hot experimental facility scheduled for operation in 1988.

- 2630 (ORNL/TM--5969) UNIFIED CREEP-

- 2634 (ORNL/TM--5995) ERRORS IN MEASURING LEVEL AND DENSITY WITH PURGED DIP TUBES. Bradley, N.C.; Murray, W.P.; Googe, J.M. (Oak Ridge National Lab., TN (USA)). Nov 1977. Contract W-7405-ENG-26. 21p. AT.
- Measurement of liquid level and density and dip-tube sensors was evaluated by analysis and experiment. A model was developed to predict the characteristic error caused by the purge-gas bubble (dip-tube bias). Experimental results are presented to support these predictions and are also compared with data reported in the literature. It was shown that properly designed and calibrated dip tubes do not make significant contributions to the error in dip-tube level and density-measurement systems.
- 2635 (ORNL/TM--6020) BREEDER REACTOR SAFETY AND CORE SYSTEMS PROGRAMS. PROGRESS REPORT, APRIL--JUNE 1977. Fontana, M.H.; Wantland, J.L. (Oak Ridge National Lab., TN (USA)). 21 Nov 1977. Contract W-7405-ENG-26. 94p. AT.
- Information is presented concerning THORS analysis and data evaluation; detection methods development; THORS facility operation; materials for fuel pin simulation; Sodium Boiling Test Facility (SBTF); neutronics analysis of a disrupted core; central computerized data base for breeder reactor safety codes; model evaluation of breeder reactor radioactivity releases; breeder reactor reliability data analysis; and environmental assessment of advanced breeder reactor fuels.
- 2636 (ORNL/TM--6025) THORIUM ASSESSMENT STUDY QUARTERLY PROGRESS REPORT, THIRD QUARTER FISCAL 1977. Spiewak, I.; Bartine, D.E.; Burns, T.J.; Cleveland, J.C.; Thomas, W.E.; White, J.R. (Oak Ridge National Lab., TN (USA)). 30 Sep 1977. Contract W-7405-ENG-26. 57p. Dep. NTIS, PC A04/MF A01.
- The objective of the program described is to contribute to the ongoing assessment of the potential role of thorium fuel cycles for alleviating safeguards concerns. Scenarios include (1) no fuel recycle permitted, (2) fuel recycle permitted only in secure regions ("energy parks") with denatured (chemically non-separable) fuels only outside these regions, and (3) no limits on fuel recycle. A further objective is to provide nuclear mass balance data on HTGRs required by ERDA contractors for comparative cost-benefit studies.
- 2637 (ORNL/TM--6029) SURVEY OF AVAILABLE CREEP AND TENSILE DATA FOR ALLOY 800H. Booker, M.K.; Baylor, V.B.; Booker, B.L.P. (Oak Ridge National Lab., TN (USA)). Jan 1978. Contract W-7405-ENG-26. 178p. AT.
- Most of the transition joints in the Clinch River Breeder Reactor Plant (CRBRP) will consist of trimetallic joints employing a spool piece (about 0.3 m long) of alloy 800H between 2 1/4 Cr-1 Mo steel and austenitic stainless steel. It is therefore important that the mechanical properties of alloy 800H be well characterized and understood. This report presents a summary of the available creep and tensile data for this material, including analytical representations of behavior.
- 2638 (ORNL/TM--6081) INSPECTION OF BOILER TUBES: STATUS AND PLANS. McClung, R.W. (Oak Ridge National Lab., TN (USA)). Oct 1977. Contract W-7405-ENG-26. 11p. AT.
- Manufacturing examination and in-service inspection techniques are being developed for steam generators for the Breeder Reactor Program. The activities in the Breeder Reactor Program (BRP) to develop and apply techniques for both manufacturing and in-service inspection of sodium/water steam generators are summarized. The principal effort has been on the units for the Clinch River Breeder Reactor (CRBR).
- 2639 (ORNL/TM--6092) ADVANCED FUEL RECYCLE PROGRAM PROGRESS REPORT, JULY 1--SEPTEMBER 30, 1977. Unger, W.E. (comp.). (Oak Ridge National Lab., TN (USA)). Nov 1977. Contract W-7405-ENG-26. 128p. AT.
- This quarterly progress report on the Fuel Recycle Program covers the ongoing development program aimed at providing the technology for production-scale advanced fuels recycle. The continuing basic laboratory and hot-cell studies, engineering research, and component equipment development activities are in support of a cold integrated equipment test scheduled for 1981 and a current conceptual design of a hot experimental facility scheduled for operation in 1988.
- 2640 (ORNL/TM--6100) CONCEPTUAL STUDY FOR THE REPROCESSING OF SPENT CARBIDE AND NITRIDE FAST REACTOR FUELS IN RELATION TO GASBORNE RADIOLOGICAL RELEASES. Morse, L.E. (Oak Ridge National Lab., TN (USA)). Dec 1977. Contract W-7405-ENG-26. 64p. AT.
- A literature survey was made to develop conceptual chemical flow diagrams for the aqueous chemical reprocessing of spent carbide and nitride fast reactor fuels in order to characterize the gasborne radioactive emissions during normal operation of the chemical plant. The conceptual flow diagrams include a head-end for the preparation of the aqueous feed for solvent extraction in which the carbides and nitrides are converted to the oxides and the internal sodium bond is deactivated by voloxidation. Uranium and plutonium values are recovered and decontaminated by Purex solvent extraction technology and subsequently converted to solid oxide products by calcination. Processes for the confinement of gasborne radionuclides are described. The estimated releases of gasborne radionuclides from the reprocessing of oxide, carbide, and nitride fast reactor fuels generating 50 GW(e) year power were very similar except for ¹⁴C, which was found to be a function of the concentration and isotopic composition of nitrogen in the fuel. 79 references, 6 tables, 5 figs.
- 2641 (ORNL/TM--6106) STEADY-STATE SODIUM TESTS IN A 19-PIN FULL-LENGTH SIMULATED LMFBR FUEL ASSEMBLY: RECORD OF STEADY-STATE EXPERIMENTAL DATA FOR THORS BUNDLE 6A. Wantland, J.L.; Clapp, N.E.; Fontana, M.H.; Gnadt, P.A.; Hanus, N.; Thornton, R.H. (Oak Ridge National Lab., TN (USA)). Mar 1978. Contract W-7405-ENG-26. 331p. AT.
- The experimental data obtained during steady-state sodium operation with test bundle 6A in the Liquid-Metal Fast Breeder Reactor (LMFBR) Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility are presented. Bundle 6A was a full-length, simulated LMFBR subassembly consisting of 19 electrically heated pins contained in a hexagonal duct. The rods had outside diameters of 5.84 mm (0.230 in.) and were spaced by 1.42-mm-diam (0.056-in.) wire wraps on a 305 mm (12.0-in.) helical pitch. The gap between the bundle and its containing duct was 0.71 mm (0.028 in.), which is half the nominal pin-to-

pin spacing. The bundle duct was of 0.51-mm-thick (0.020-in.) type 316 stainless steel backed by insulation to provide low thermal inertia. The heated length was 900 mm (36 in.) with variable-pitch heater windings to produce a 1-3 axial peak-to-mean power distribution, and an unheated simulated fission gas plenum was downstream of the heated section. The primary purpose of bundle 6A steady-state tests was to establish bundle temperature distributions at various flow-power ratios with a fixed number of rods heated. These temperature distributions can be compared with those of previous tests and with those of future tests with larger bundles.

- 2642 (ORNL/TM--6128) STRESS-STRAIN BEHAVIOR OF ERNCR-3 WELD FILLER METAL UNDER STRAIN CONTROLLED CYCLIC LOADING. Baylor, V.B.; Booker, M.K.; Strizak, J.P.; Klueh, R.L. (Oak Ridge National Lab., TN (USA)). Jan 1978. Contract W-7405-ENG-26. 39p. AT.

Weld filler metal ERNCR-3 is expected to be used in the transition joints of the steam generators in breeder reactors. Design and analysis of these joints require consideration of the mechanical properties of the materials involved. This study suggests analytical representations for the cyclic stress-strain response of ERNCR-3 as observed during strain-controlled continuous-cycling fatigue tests. Specifically, areas to be discussed include cyclic hardening and softening during testing, development of cyclic stress-strain curves from stress-strain amplitude coordinates, and development of bilinear representations of cyclic stress-strain behavior as recommended for inelastic analysis by current interim inelastic design rules. Those rules were originally developed for austenitic steels, but were extended here to provide a framework within which to analyze the behavior of ERNCR-3. 33 figures, 4 tables.

- 2643 (ORNL/TM--6158) BREEDER REACTOR SAFETY AND CORE SYSTEMS PROGRAMS. PROGRESS REPORT, JULY--SEPTEMBER 1977. Fontana, M.H.; Wantland, J.L. (Oak Ridge National Lab., TN (USA)). Mar 1978. Contract W-7405-ENG-26. 120p. AT.

Topics discussed are program coordination, Thermal-Hydraulic Out-of-Reactor Safety (THORS) analysis and data evaluation, THORS facility operation, materials for fuel pin simulation, the sodium boiling test facility, neutronics analysis of a disrupted core, central computerized data base for breeder reactor safety codes, model evaluation of breeder reactor reactivity releases, breeder reactor reliability data analysis center, and environmental assessment of breeder reactor fuels.

- 2644 (ORNL/TM--6170) ROOM-TEMPERATURE INELASTIC TEST OF A NOZZLE-TO-SPHERICAL-SHELL MODEL SUBJECTED TO PRESSURE AND MOMENT LOADING. Gwaltney, R.C.; Richardson, M.; Battiste, R.L. (Oak Ridge National Lab., TN (USA)). Apr 1978. Contract W-7405-ENG-26. 36p. AT.

Portions of document are illegible. The measured room-temperature elastic-plastic response is presented for a nozzle attached to a spherical shell that was subjected to an internal pressure loading and a cyclic moment applied to the nozzle. The test was performed to qualify the facility and provide experimental benchmark problem data suitable for assessing inelastic analysis methods and validating structural analysis computer programs. The nozzle-to-sphere model is representative of geometries encountered in

design because it exhibits the essential features of inelastic behavior of nozzle attachments in pressure vessels. The specimen was machined from a well-characterized heat of type 304 stainless steel, and all tests were performed at a temperature of 21°C (70°F). Test results are presented in terms of loads and strains that typify the overall structural behavior.

- 2645 (ORNL/TM--6228) LONAC: A COMPUTER PROGRAM TO INVESTIGATE SYSTEMS DYNAMICS UNDER CONDITIONS OF LOW FORCED FLOW AND NATURAL CONVECTION. Ribando, R.J. (Oak Ridge National Lab., TN (USA)). 16 Mar 1978. Contract W-7405-ENG-26. 47p. AT.

Portions of document are illegible. LONAC (LOw flow and NATural Convection), a fast-executing code for studying systems dynamics under conditions of low forced flow and natural convection, is discussed. Although written specifically for the THORS loop at ORNL, it can easily be adapted to other thermal-hydraulic facilities. The components included in the code are electrically heated fuel pin simulators, an expansion tank, a pump, valves, a heat dump, and a bypass line. The one-dimensional, single-phase conservation equations are written assuming density is solely a function of temperature, thus eliminating acoustic effects.

- 2646 (ORNL/TM--6234) PLAN FOR THE DETERMINATION OF MECHANICAL AND THERMAL PROPERTIES OF CONCRETES TO TEMPERATURES OF 621°C (1150°F). Oland, C.B.; Robinson, G.C.; Dougan, J.R.; Callahan, J.P. (Oak Ridge National Lab., TN (USA)). Sep 1978. Contract W-7405-ENG-26. 63p. AT.

A detailed description of a testing program designed to define the variations in physical properties of limestone aggregate structural concrete and lightweight insulating concrete exposed to sustained heating at discrete temperatures up to 621°C (1150°F) is given. A total of 200 specimens will be tested to identify basic trends in the effect of temperature on unconfined compressive strength, shear strength, rebar bond strength, sustained loading (creep) behavior and thermal properties consisting of expansion, conductivity and diffusivity.

- 2647 (ORNL/TM--6285) INFLUENCE OF SMALL AMOUNTS OF NIOBIUM ON MECHANICAL AND CORROSION PROPERTIES OF TYPE 304 STAINLESS STEEL. Sikka, V.K.; Moorhead, A.J.; Brinkman, C.R. (Oak Ridge National Lab., TN (USA)). Jun 1978. Contract W-7405-ENG-26. 77p. AT.

Creep and creep-rupture properties of several experimental heats of type 304 stainless steel containing 20, 50, 100, 200, 500, 700, and 1000 ppm (by weight) Nb were investigated in the temperature range 482 to 649°C. Resistance to intergranular corrosion of these heats was also studied. Results of the tests on the experimental heats were compared with the creep properties of 20 commercial heats of type 304 stainless steel and data from the literature on several heats of type 347 (4000 to 8000 ppm Nb) stainless steel. Results on experimental heats were used to check that heat-to-heat variations in creep properties of 20 commercial heats of type 304 stainless steel are caused by variations in carbon, nitrogen, grain size, and for the most part by residual niobium (10 to 200 ppm). The time to rupture increased and minimum creep rate decreased with increasing niobium content until a saturation level was attained at about 500 to 1000 ppm. Type 304 stainless steel containing 500 ppm Nb

offers superior creep and creep-rupture properties with improved creep ductility and a factor of approximately 5 better intergranular corrosion resistance (in boiling CuSO_4 solution) than type 304 stainless steel (20 ppm Nb). Type 304 containing 500 to 1000 ppm Nb exhibits a weldability similar to that of type 304 stainless steel.

- 2648 (ORNL/TM--6287) STABILITY OF IODIC ACID UNDER IODOX INTERIM STORAGE CONDITIONS. Clark, W.E.; Mailen, J.C. (Oak Ridge National Lab., TN (USA)). Aug 1978. Contract W-7405-ENG-26. 21p. AT.

Small-scale irradiation experiments indicate that little, if any, volatilization of I_2 will occur from irradiated solutions of iodic acid in azetropic HNO_3 , as long as the solutions continue to boil vigorously. This conclusion is confirmed by experiments in which the radiolytic generation of nitrite was simulated by introducing N_2O_5 gas into a boiling solution of iodic acid in HNO_3 . Free I_2 formed rapidly in cooler solutions. It appears that only a minimum sparge will be required to prevent volatilization of iodine from the Iodex storage tank, but that a rapid buildup of free iodine may occur if boiling is allowed to cease. The minimum G-value for generation of HNO_2 in the boiling storage solution was found to be approximately 9.7. The actual G-value is almost certainly somewhat higher, with 14.6 as the probable maximum.

- 2649 (ORNL/TM--6288) BREEDER REACTOR SAFETY AND CORE SYSTEMS PROGRAMS PROGRESS REPORT, OCTOBER--DECEMBER 1977. Fontana, M.H.; Wantland, J.L. (Oak Ridge National Lab., TN (USA)). Jun 1978. Contract W-7405-ENG-26. 107p. AT.

This report covers progress made during October--December 1977 by the Breeder Reactor Safety and Core Systems Programs at the Oak Ridge National Laboratory. Topics discussed are program coordination, Thermal-Hydraulic Out-of-Reactor Safety (THORS) analysis and data evaluation, THORS facility operation, Sodium Boiling Test Facility (SBTF), materials for fuel pin simulation, neutronics of a disrupted core, environmental assessment of breeder reactor fuels, model evaluation of breeder reactor radioactivity releases, Nuclear Safety Information Center, central computerized data base for breeder reactor safety codes, and breeder reactor reliability data analysis center.

- 2650 (ORNL/TM--6296) IN-REACTOR DEFORMATION AND FRACTURE OF AUSTENITIC STAINLESS STEELS. Bloom, E.E.; Wolfer, W.G. (Oak Ridge National Lab., TN (USA)). Jun 1978. Contract W-7405-ENG-26. 26p. AT.

An experimental technique for determining in-reactor fracture strain was developed and demonstrated. Differential swelling between a sample holder and a test specimen with a lower swelling rate produced uniaxial deformation. In-reactor deformations of 0.7 to 2.1% were achieved in type 304 stainless steel previously irradiated to fluences up to 8.8×10^{22} n/m² without fracture. These strains are significantly higher than found in postirradiation creep-rupture tests on similar samples. From the measured strain values and published irradiation creep data and correlations, the stress levels during the irradiation were calculated. On the basis of previous postirradiation creep-rupture results, many of the samples that did not fail would be predicted to fail. Thus we conclude that the in-reactor rupture life is longer than predicted

by postirradiation tests. Strain in a fractured sample was estimated to be less than 3.8%, and the in-reactor fractures were intergranular--the same fracture mode as found in postirradiation tests. Irradiation creep may relax stresses at crack tips and sliding boundaries, thus retarding the initiation and/or growth of cracks and leading to longer rupture lives in-reactor. However, the very high ductility or superplastic behavior predicted by the strain rate sensitivity of irradiation creep is not achieved because of the eventual interruption of the deformation process by grain boundary fracture.

- 2651 (ORNL/TM--6306) ADVANCED FUEL RECYCLE PROGRAM PROGRESS REPORT, JANUARY 1 TO MARCH 31, 1978. Unger, W.E. (comp.). (Oak Ridge National Lab., TN (USA)). May 1978. Contract W-7405-ENG-26. 135p. AT.

This quarterly progress report on the Fuel Recycle Program covers the ongoing development program aimed at providing the technology for production-scale advanced fuels recycle. The continuing basic laboratory and hot-cell studies, engineering research, and component equipment development activities are in support of a cold integrated prototype equipment test scheduled for 1981 and a current conceptual design of a hot experimental facility planned for operation in the early 1990s.

- 2652 (ORNL/TM--6309) PRELIMINARY NEUTRONIC STUDY OF ACTINIDE TRANSMUTATION IN A FAST REACTOR. Williams, M.L.; McAdoo, J.W.; Flanagan, G.F. (Oak Ridge National Lab., TN (USA)). Aug 1978. Contract W-7405-ENG-26. 33p. Dep. NTIS, PC A03/HF A01.

Preliminary results are cited from a neutronic study of actinide recycle in a fast reactor, using an LMFBR model. The work was performed as part of a Chemical Technology Division program for assessing the merit of actinide transmutation as a means of waste disposal. Calculations were performed by coupling two-dimensional diffusion theory calculations with point-depletion recycle calculations. Cross sections were obtained from a "preliminary" version of ENDF/B-V. Results are given concerning the recycled-actinide equilibrium condition and the waste-actinide destruction rate. The effect of adding U-238 diluent to reduce power peaking is examined parametrically. The heterogeneous effects of placing an actinide target assembly in a fuel lattice are also examined.

- 2653 (ORNL/TM--6343) SIMBD: A SIMPLE BOILING MODEL OF THE RESPONSE OF A SIMULATED SODIUM-COOLED BREEDER REACTOR SUBASSEMBLY TO AN UNDERCOOLING TRANSIENT. Garrison, P.W. (Oak Ridge National Lab., TN (USA)). Sep 1978. Contract W-7405-ENG-26. 59p. AT.

Portions of document are illegible. A highly simplified thermal-hydraulic code, SIMBD, has been developed to investigate the influence of the fuel-assembly duct wall on the results of a series of boiling tests in the THORS facility at ORNL. The code is based on an implicit solution to a single-channel axial energy transport equation coupled through a film coefficient to an explicit solution to a multiple-radial-node conduction model. A quasi-steady approximation of the momentum equation is used in conjunction with the Martinelli-Lockhart two-phase flow correlation. Predictions made by the model are compared with experimental results from the THORS bundle 6A test series.

- 2654 (ORNL/TM--6373) FLUID MECHANICS MODEL TO ESTIMATE THE LEAKAGE OF INCOMPRESSIBLE FLUIDS THROUGH LABYRINTH SEALS. Han, J.T. (Oak Ridge National Lab., TN (USA)). Aug 1978. Contract W-7405-ENG-26. 29p. AT.

An analytical model for estimating the leakage of incompressible fluids through straight labyrinth seals has been described. Recent developments in theoretical and experimental studies of recirculating flow inside rectangular cavities and of flow in a channel have been incorporated in the model to solve a practical problem. The approach used in this model is different from those of previous ones for compressible fluids in which the gas-dynamics laws with some empirical coefficients determined from gas data were used. The seal leakages predicted by the model are in reasonable agreement with these measured indirectly in sodium and those obtained for gas at low Mach number and small pressure variations. It can be concluded from this study that the labyrinth seal formed with cavities has less leakage than the one formed with threads if all other conditions are the same. Furthermore, the fluid leakage through straight labyrinth seals (for both cavity and thread types) is strongly dependent upon the seal clearance and pressure drop.

- 2655 (ORNL/TM--6393) INSTRUMENTATION AND CONTROL DEVELOPMENT AND DESIGN PHILOSOPHY FOR ADVANCED FUEL RECYCLE. Hamel, W.R.; Bradley, N.C.; Jansen, J.M. Jr. (Oak Ridge National Lab., TN (USA)). Aug 1978. Contract W-7405-ENG-26. 39p. AT.

This document provides a definition and an explanation of the instrumentation and controls philosophy being utilized in the Advanced Fuel Recycle Program (AFRP). The general objectives and assumptions used in defining the philosophy are based on overall AFRP objectives. The philosophy presents architectural concepts, functional capabilities, performance requirements, safeguards factors, and implementation techniques. A brief summary of current development activities is also presented.

- 2656 (ORNL/TM--6408) MODEL OF A FISSION COUNTER SYSTEM FOR OPTIMIZING PERFORMANCE AT HIGH GAMMA DOSE RATES. Pare, V.K. (Oak Ridge National Lab., TN (USA)). Nov 1978. Contract W-7405-ENG-26. 44p. Dep. NTIS, PC A03/MF A01.

A mathematical model has been developed for predicting the ability of a fission counter to operate at a high gamma dose rate. There are several potential applications in reactor programs for detectors that can accurately measure neutron fluxes in the range 0.1 to 1.0 nv(th) while subjected to gamma dose rates of 10^4 to more than 10^6 R/hr. Fission counters for such service have up to now been designed on the basis of laborious testing of prototypes combined with extrapolation from existing designs. The objective of this work is to make it possible to arrive at an optimum design without expensive testing. In this model, the complete system of counter, cables, preamplifier, main amplifier, filters, and discriminator is considered, including the effects of interaction of counter capacitance with cable and preamplifier impedance. Electrical noise theory is used to calculate the frequency spectrum of the interfering signal generated by pileup of gamma pulses, to add to it a contribution from electronic noise, and to obtain the rate at which the total noise signal triggers the discriminator. The shapes and widths of neutron and gamma pulses are calculated from the electron drift velocity and

electrode spacing.

- 2657 (ORNL/TM--6411) ASSESSMENT OF NORMALIZED AND TEMPERED VERSUS ANNEALED OR ISOTHERMALLY ANNEALED 2-1/4 CR--1 MO STEEL. Klueh, R.L.; Booker, M.K. (Oak Ridge National Lab., TN (USA)). Aug 1978. Contract W-7405-ENG-26. 43p. AT.

Commercially, 2 1/4 Cr-1 Mo steel is used in various heat-treated conditions. Present standards for breeder reactor steam generators specify that the steel will be used only in an annealed or isothermally annealed condition. Future breeder reactor steam generators may offer an incentive to use a normalized-and-tempered steel. The property differences between the normalized-and-tempered and annealed or isothermally annealed conditions are assessed to determine the advisability of using the normalized-and-tempered material for breeder reactor steam generators.

- 2658 (ORNL/TM--6439) BREEDER REACTOR SAFETY AND CORE SYSTEMS PROGRAMS. PROGRESS REPORT, JANUARY--MARCH 1978. Fontana, M.H.; Wantland, J.L. (Oak Ridge National Lab., TN (USA)). Sep 1978. Contract W-7405-ENG-26. 91p. AT.

Portions of document are illegible.

Progress made during January--March 1978 by the Breeder Reactor Safety and Core Systems Programs at the Oak Ridge National Laboratory is reported. Topics discussed are program coordination, Thermal-Hydraulic Out-of-Reactor Safety (THORS) analysis and data evaluation, THORS Facility operation, Sodium Boiling Test Facility (SBTF), materials for fuel pin simulation, neutronics of a disrupted core, environmental assessment of breeder reactor fuels, model evaluation of breeder reactor radioactivity releases, Nuclear Safety Information Center, central computerized data base for breeder reactor safety codes, and breeder reactor reliability data analysis center.

- 2659 (ORNL/TM--6464) INTERIM ANALYTICAL DESCRIPTION OF THE CREEP AND CREEP-RUPTURE BEHAVIOR OF ERN1Cr-3 WELD METAL. Booker, M.K.; Klueh, R.L. (Oak Ridge National Lab., TN (USA)). Nov 1978. Contract W-7405-ENG-26. 58p. AT.

Filler metal of AWS A5.14 Class ERN1Cr-3 (67% Ni--20% Cr--3% Mn--2.5% Nb), often known as Inconel 82, is conventionally used for austenitic-to-ferritic dissimilar-metal transition weld joints. For example, this filler metal is proposed for use in the transition joints for the steam generator system of the Clinch River Breeder Reactor Plant. Recent efforts to develop analytical descriptions of the creep and creep-rupture behavior of ERN1Cr-3 weld metal are described. These results are compared with previous analyses of the behavior of 2 1/4 Cr--1 Mo steel and alloy 800H, the two base metals that would be joined by the ERN1Cr-3 filler metal in one of the proposed CRBRP transition joints.

- 2660 (ORNL/TM--6497) DISCRETE ORDINATE ANALYSIS OF THE TSF EXPERIMENT ON NEUTRON STREAMING IN A CRBR PROTOTYPIC COOLANT PIPE CHASEWAY. Maerker, R.E. (Oak Ridge National Lab., TN (USA)). Nov 1978. Contract W-7405-ENG-26. 64p. AT.

The results are summarized of calculations of neutrons streaming through a prototypic multilegged CRBR coolant pipe chaseway performed with the two-dimensional discrete ordinates code DOT 3.5, together with a

pecially written auxiliary routine that transformed the angular fluxes transmitted through one leg into an approximation source for the calculation of the next leg. Comparisons with measured fluxes at intermediate points along the duct indicate an approximate factor-of-two accuracy in the S_{sub}/n calculations using a recommended procedure for geometric modeling of each leg. The accuracy of an alternate geometric approximation was assessed and found to be generally inferior to that of the recommended procedure.

- 2661 (ORNL/TM--6498) STEADY-STATE SODIUM TESTS IN A 19-PIN INTERNALLY GUARD-HEATED SIMULATED LMFBR FUEL ASSEMBLY WITH A SIX-CHANNEL INTERNAL BLOCKAGE: RECORD OF EXPERIMENTAL DATA FOR THORS BUNDLE 3C. Hanus, N.; Nelson, W.R.; Clapp, N.E.; Fontana, M.H.; Gnadt, P.A.; Thornton, R.H.; Wantland, J.L. (Oak Ridge National Lab., TN (USA)). Feb 1979. Contract W-7405-ENG-26. 379p. AT.

The report is a record of experimental data obtained during steady-state sodium operation with test bundle 3C in the Liquid-Metal Fast Breeder Reactor (LMFBR) Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility. Bundle 3C consisted of 19 heated pins [fuel pin simulators (FPSs)] surrounded by 12 heated edge pins installed in a stainless steel duct. The pins had outside diameters of 5.84 mm and were spaced by 1.42-mm-dia wire wraps on a 305-mm pitch, the same as for the proposed Clinch River Breeder Reactor (CRBR). A 6.4-mm stainless steel plate was attached to the central pin and blocked the six central channels 381 mm downstream from the start of the heated section. The uniformly heated length of the bundle was 533 mm. Hollow pins 1.6 m long, which simulate the fission gas plenum of a reactor, were attached to the downstream end of the 19 central FPSs. The purpose of the bundle 3C steady-state tests was to determine the thermal-hydraulic effects of the internal blockage at various flow-power ratios with a fixed number of pins heated. The report includes a description of the test facility and instrumentation, data reduction procedures, and the test data from experiments completed in this series.

- 2662 (ORNL/TM--6522) FUEL CYCLE COST STUDIES: FABRICATION, REPROCESSING, AND REFABRICATION OF LWR, SSCR, HWR, LMFBR, AND HTGR FUELS. Olsen, A.R.; Judkins, R.R.; Carter, W.L.; Delene, J.G. (Oak Ridge National Lab., TN (USA)). Mar 1979. Contract W-7405-ENG-26. 41p. Dep. NTIS, PC A03/MF A01.

The comparative analysis of power generation costs for the various reactor cycles that is being performed in the Nonproliferation Alternative Systems Assessment Program (NASAP) and the International Nuclear Fuel Cycle Evaluation (INFCE) requires that the costs associated with processing of fuel materials for use in these cycles be estimated. The study described here provided unit cost estimates for the fabrication, reprocessing, and refabrication of a variety of fuels for several reactor systems. We examined in detail the facility requirements and operations to estimate capital and operating costs. Unit processing cost determinations were based on a cash flow analysis technique in which income from sales over the life of each facility was equated to the total capital and operating expenses of that facility plus a specified return on equity investment. The effects of plant capacities were determined by application of scaling factors to individual components of

the reference plant costs. Capital and operating costs were estimated for 21 reactor and fuel cycle combinations. Based on these estimates, unit costs were determined for fabrication, reprocessing, and refabrication of the fuels. In each instance, the effect of plant capacities on unit costs associated with the processing of fuels was determined. All costs were based on mature industries, and first-of-a-kind costs were not included. Unit cost determinations were based on three financing techniques, which included government financing, typical industrial financing, and high-risk industrial financing. The unit costs recommended for the comparative analysis of power generation costs are those associated with the economic assumptions of a typical industry.

- 2663 (ORNL/TM--6531) MICROCOMPUTER-CONTROLLED ULTRASONIC DATA ACQUISITION SYSTEM. Simpson, W.A. Jr. (Oak Ridge National Lab., TN (USA)). Nov 1978. Contract W-7405-ENG-26. 55p. Dep. NTIS, PC A04/MF A01.

The large volume of ultrasonic data generated by computer-aided test procedures has necessitated the development of a mobile, high-speed data acquisition and storage system. This approach offers the decided advantage of on-site data collection and remote data processing. It also utilizes standard, commercially available ultrasonic instrumentation. This system is controlled by an Intel 8080A microprocessor. The MCS80-SDK microcomputer board was chosen, and magnetic tape is used as the storage medium. A detailed description is provided of both the hardware and software developed to interface the magnetic tape storage subsystem to Biomation 8100 and Biomation 805 waveform recorders. A boxcar integrator acquisition system is also described for use when signal averaging becomes necessary. Both assembly language and machine language listings are provided for the software.

- 2664 (ORNL/TM--6553) SODIUM BOILING IN A FULL-LENGTH 19-PIN SIMULATED FUEL ASSEMBLY (THORS BUNDLE 6A). Ribando, R.J.; Clapp, N.E.; Fontana, M.H. (Oak Ridge National Lab., TN (USA)). Jan 1979. Contract W-7405-ENG-26. 117p. AT.

Sodium boiling in liquid-metal fast breeder reactor (LMFBR) fuel subassemblies was investigated using an electrically heated 19-pin bundle in the Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility. Results of 6 of some 60 tests are presented in detail. From these results and from analytical results also discussed, it appears that increased thermal inertia due to sodium in the insulation surrounding the pin assembly contributed to incoherent (i.e., non-one-dimensional) boiling and delayed dryout.

- 2665 (ORNL/TM--6558) BREEDER REACTOR SAFETY AND CORE SYSTEMS PROGRAMS. PROGRESS REPORT, APRIL--JUNE 1978. Fontana, M.H.; Wantland, J.L. (Oak Ridge National Lab., TN (USA)). Nov 1978. Contract W-7405-ENG-26. 86p. AT.

This report covers progress made during April--June 1978 by the Breeder Reactor Safety and Core Systems Programs at Oak Ridge National Laboratory. Topics discussed are program coordination, Thermal-Hydraulic Out-of-Reactor Safety (THORS) analysis and data evaluation, THORS facility operation, Sodium Boiling Test Facility (SBTF), material for fuel pin simulation, neutronics of a disrupted core, environmental assessment of alternate FBR

fuels, model evaluation of breeder reactor radioactivity releases, Nuclear Safety Information Center, central computerized data base for breeder reactor safety codes, and breeder reactor reliability data analysis center.

- 2666 (ORNL/TM--6568) HYDROGEN RETENTION OF (U,Pu)O/SUB 2-Y/. Pechin, W.H.; Bradley, R.A.; Sease, J.D. (Oak Ridge National Lab., TN (USA)). Dec 1978. Contract W-7405-ENG-26. 17p. AT.

An improved process for sintering (U,Pu)O/sub 2-y/ fast breeder reactor fuel was developed. The process resulted in lower gas content in the as-fabricated fuel. One of the specifications for such fuel is that the amount of gas released by vacuum extraction at 1600°C be less than 90 cm³/kg. In the preparation of (U,Pu)O/sub 2-y/ for irradiation testing, pellets with densities less than 90% of theoretical could release 10 to 20 times the maximum amount of gas permitted. Thus, an improved process was required. Several approaches were used to investigate what gas was being retained by the fuel and the retention mechanism. First, fuel was cooled from the sintering temperature in different gas atmospheres, and the quantity and type of retained gas were determined. Gas analyses of sintered material as a function of extraction temperature were also informative. Gas release measurements on fuels of varying degrees of oxygen stoichiometry helped define the retention mechanism. The retained gas was found to be primarily hydrogen, which had combined with the fuel during cooling after hydrogen reduction. The amount of retained hydrogen was greatest for most highly reduced and porous fuel.

- 2667 (ORNL/TM--6577) ADVANCED FUEL RECYCLE PROGRAM. PROGRESS REPORT, JULY 1--SEPTEMBER 30, 1978. Unger, W.E. (comp.). (Oak Ridge National Lab., TN (USA)). Dec 1978. Contract W-7405-ENG-26. 113p. AT.

This quarterly progress report on the Fuel Recycle Program covers the ongoing development program aimed at providing the technology for production-scale advanced fuels recycle. The continuing basic laboratory and hot-cell studies, engineering research, and component equipment development activities are in support of a cold Integrated Equipment Test scheduled for 1981 and a current conceptual design of a Hot Experimental Facility planned for operation in the early 1990's.

- 2668 (ORNL/TM--6582) NONLINEAR PLASTICITY MODEL FOR STRUCTURAL ALLOYS AT ELEVATED TEMPERATURE. Robinson, D.N. (Oak Ridge National Lab., TN (USA)). Nov 1978. Contract W-7405-ENG-26. 33p. AT.

A nonlinear, time-independent plasticity model is presented which incorporates some aspects of both isotropic and kinematic hardening. The model characterizes a material with limited memory, i.e., in the sense that part of the deformation history as recorded in the internal dislocation structure is erased at stress reversals. This feature ensures that the predicted response eventually reaches a limit cycle under cyclic stressing, even in the presence of creep and relaxation. The model is intended as a candidate for replacing the nonlinear model now residing in Sect. 4.3.6 of RDT Standard F9-5T.

- 2669 (ORNL/TM--6600) ELECTRON-BEAM WELDING OF FUEL PIN SIMULATORS TO A TUBESHEET. Gnadt,

P.A.; King, J.F. (Oak Ridge National Lab., TN (USA)). Dec 1978. Contract W-7405-ENG-26. 24p. AT.

The Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility, a large-scale sodium flow facility at the Oak Ridge National Laboratory (ORNL), was used to test simulated Liquid-Metal Fast Breeder Reactor (LMFBR) fuel subassemblies for safety-related occurrences. These subassemblies are composed of bundles of electrically heated rods with the same linear power density and external configuration as LMFBR fuel rods. This report describes the electron-beam (EB) welding procedures developed for joining these rods [known as fuel pin simulators (FPSs)] to a tubesheet as part of the fabrication of the complex test assemblies.

- 2670 (ORNL/TM--6640) NUCLEAR FUEL FABRICATION AND REFABRICATION COST ESTIMATION METHODOLOGY. Judkins, R.R.; Olsen, A.R. (Oak Ridge National Lab., TN (USA)). Nov 1979. Contract W-7405-ENG-26. 70p. Dep. NTIS, PC A04/MF A01.

The costs for construction and operation of nuclear fuel fabrication facilities for several reactor types and fuels were estimated, and the unit costs (prices) of the fuels were determined from these estimates. The techniques used in estimating the costs of building and operating these nuclear fuel fabrication facilities are described in this report. Basically, the estimation techniques involve detailed comparisons of alternative and reference fuel fabrication plants. Increases or decreases in requirements for fabricating the alternative fuels are identified and assessed for their impact on the capital and operating costs. The impact on costs due to facility size or capacity was also assessed, and scaling factors for the various capital and operating cost categories are presented. The method and rationale by which these scaling factors were obtained are also discussed. By use of the techniques described herein, consistent cost information for a wide variety of fuel types can be obtained in a relatively short period of time. In this study, estimates for 52 fuel fabrication plants were obtained in approximately two months. These cost estimates were extensively reviewed by experts in the fabrication of the various fuels, and, in the opinion of the reviewers, the estimates were very consistent and sufficiently accurate for use in overall cycle assessments.

- 2671 (ORNL/TM--6679) ULTRASONIC EVALUATION OF DISSIMILAR-METAL TRANSITION JOINT WELDS. Smith, J.H. (Oak Ridge National Lab., TN (USA)). Apr 1977. Contract W-7405-ENG-26. 48p. AT.

Ultrasonic techniques for the nondestructive examination of the dissimilar-metal transition joints designed for the Clinch River Breeder Reactor were studied. This included determining the acoustical properties of the various materials involved and evaluating the relative merits of several examination techniques and associated parameters. Using the results of these studies, ultrasonic inspection methods that can be used to evaluate 12.7- and 25.4-mm-thick dissimilar-metal welds in transition joints in the as-manufactured (machined) condition were developed. A 2:1 signal-to-noise ratio notches whose depths are 8% of the weld thickness at any of the four weld root interfaces and in the weld root volume of both welds can be detected. Better sensitivity can be obtained if multiple test frequencies are used and if only the root areas of the welds and weld interfaces are considered.

- 2672 (ORNL/TM--6679/R1) ULTRASONIC EVALUATION OF DISSIMILAR-METAL TRANSITION JOINT WELDS. Smith, J.H. (Oak Ridge National Lab., TN (USA)). Jun 1979. Contract W-7405-ENG-26. 48p. AT.

We have studied ultrasonic techniques for the nondestructive examination of the dissimilar-metal transition joints designed for the Clinch River Breeder Reactor. This included determining the acoustical properties of the various materials involved and evaluating the relative merits of several examination techniques and associated parameters. Using the results of these studies we developed ultrasonic inspection methods that can be used to evaluate 12.7- and 25.4-mm-thick dissimilar-metal welds in transition joints in the as-manufactured (machined) condition. We can routinely detect with a 2:1 signal-to-noise ratio notches whose depths are 8% of the weld thickness at any of the four weld root interfaces and in the weld root volume of both welds. Better sensitivity can be obtained if multiple test frequencies are used and if only the root areas of the welds and weld interfaces are considered.

- 2673 (ORNL/TM--6688) FUEL PIN SIMULATORS FOR SODIUM BOILING TESTS IN THE THORS FACILITY. Gnadt, P.A.; MacPherson, R.E.; McCulloch, R.W. (Oak Ridge National Lab., TN (USA)). Apr 1979. Contract W-7405-ENG-26. 65p. AT.

The Breeder Reactor Program (BRP) at Oak Ridge National Laboratory (ORNL) requires high-performance, electrically heated fuel pin simulators (FPSs) to duplicate the configuration, heat flux profiles, and operational capabilities of nuclear fuel pins. These FPSs have been assembled into 19-pin bundles and operated to temperature levels sufficient to cause sodium boiling in the Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility at ORNL. This report describes the procurement, assembly, and evaluation of the FPSs and their subsequent operation in THORS as bundles 6A and 3C. The two test bundles operated for over 11,000 hr (total) in sodium at temperatures between 316°C (600°F) and 1010°C (1850°F). Over 60 tests were run to conditions where sodium boiling occurred. Boiling was usually sustained for a period of 10 to 30 sec. Several of the tests resulted in clad dryout (unstable) conditions. The bundle 3C FPSs contained a uniform-pitch Nichrome V heating element, and those in bundle 6A contained a constant-width, variable-pitch Pt-8% W heating element, which gave a chopped-cosine heat flux pattern to the FPS. A complete description of the two FPS designs is presented.

- 2674 (ORNL/TM--6698) BREEDER REACTOR SAFETY AND CORE SYSTEMS PROGRAMS. PROGRESS REPORT, JULY--SEPTEMBER 1978. Fontana, M.H.; Wantland, J.L. (Oak Ridge National Lab., TN (USA)). Feb 1979. Contract W-7405-ENG-26. 92p. AT.

Topics discussed are program coordination, THORS analysis and data evaluation, THORS Facility operation, SBTf, material for fuel pin simulation, neutronics of a disrupted core, environmental assessment of alternate FBR fuels, model evaluation of breeder reactor radioactivity releases, NSIC, central computerized data base for breeder reactor safety codes, and breeder reliability data analysis center. The completed THORS Bundle 3C (8% of flow area blocked) boiling tests are summarized, and typical data are presented. Preliminary analysis indicates that no dryout and no bulk flow instability occurred during

sodium boiling. The SAS 3A computer code was used to study the interaction between two heated bundles operated in parallel with a bypass flow channel. These results indicate that bundle interaction depends on bundle inlet flow impedance. The THORS Facility has operated for a total of 26,400 hr. Test bundles have been operated at varying power levels for approx. 2300 hr; sodium boiling tests have been conducted in four bundles. The boiling propagation tests for Bundle 3C have been completed. The facility has been shut down for modification and for installation of the 61-pin Bundle 9. The piping to the new 2.0-MW heat dump will also be installed, giving the facility a full 2.0-MW(e) input and heat-removal capability. Assembly of Bundle 9 is 90% complete. The SBTf became operational during this reporting period. Approximately 2 hr of boiling time was logged before distortion of the test section forced shutdown for refurbishment. Work on the quasi-static Monte Carlo Code SPQR has been halted. Present efforts in this task are directed to safety sensitivity analysis. Two reports on environmental assessment of alternate FBR fuels were submitted during this period for publication. Sites for weather monitoring equipment in the Lemhi Pass thorium resource region were selected following an aerial and ground survey.

- 2675 (ORNL/TM--6744) CORROSION BEHAVIOR OF LMFBR FUEL CLADDING IN SIMULATED WATER POOL STORAGE CONDITIONS. Wu, P.C.S. (Oak Ridge National Lab., TN (USA)). May 1979. Contract W-7405-ENG-26. 59p. AT.

The corrosion behavior of sodium-exposed prototypic LMFBR fuel cladding was investigated under simulated water pool storage conditions for breeder reactor spent fuels. As an extension of a previous scoping study on a similar subject, the current investigation has not only generated additional data, which has enhanced the statistical confidence of the results obtained in the scoping study, but baseline information has also been provided for the corrosion behavior of irradiated cladding specimens. Water exposure of the cladding specimens was conducted in simulated breeder reactor water pool storage conditions for periods up to 12 months at 82°C. No observable material deterioration or failure occurred among the hot-leg cladding specimens (exposed in the corrosion region of a dynamic sodium loop).

- 2676 (ORNL/TM--6745) DATA MANAGEMENT SYSTEM FOR THE THERMAL-HYDRAULIC OUT-OF-REACTOR SAFETY FACILITY COMPUTER-CONTROLLED DATA ACQUISITION SYSTEM CONFIGURATION FOR BUNDLE 3C OPERATION. Clapp, N.E. (Oak Ridge National Lab., TN (USA)). Feb 1979. Contract W-7405-ENG-26. 51p. AT.

The Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility Data Acquisition System (DAS) consists of a PDP-8E computer with a 32,000-word memory, a 1.6-million-word disk, a 7-track magnetic tape for data storage, a teletype, a cathode-ray-tube (CRT) display, a character printer, a 256-channel multiplexer, gain selectable amplifiers (one per box), and an analog-to-digital (A/D) converter. The data system has the capability of sampling, checking out-of-limit situations, and storing data at a rate of 10,000 channels/sec. The data system under computer control provides the capability of near-real-time data analysis and quick-look displays of test data. The software system is designed to provide the flexibility required to meet the demands of the facility research program. The software system consists of two

levels: (1) the interpreter, which transforms program statements into machine language statements; and (2) the collection of program statements required to perform a given data acquisition or calculation task. The data stored on tape by this system are not directly usable on the IBM 360 computers. The FORTRAN programs required to transform and convert the data tape into a compatible engineering unit data tape are discussed.

- 2677 (ORNL/TM--6778) COMPARISON OF NUMERICAL RESULTS WITH EXPERIMENTAL DATA FOR SINGLE-PHASE NATURAL CONVECTION IN AN EXPERIMENTAL SODIUM LOOP (THORS BUNDLE 6A). Ribando, R.J.; Clapp, N.E.; Fontana, M.H.; Hanus, N.; Nelson, W.R.; Wantland, J.L. (Oak Ridge National Lab., TN (USA)). Sep 1979. Contract W-7405-ENG-26. 88p. AT.

A comparison is made between computed results and experimental data for single-phase natural convection in an experimental sodium loop. The tests were conducted in the Thermal-Hydraulic Out-of-Reactor Safety (THORS) facility. Heat generation in the 19 pin assembly during these tests was typical of decay heat levels. Tests were conducted both with zero initial forced flow and with a small initial forced flow. The bypass line was closed in most tests, but open in one. The computer code used to analyze these tests [LONAC (LOW flow and NATURAL Convection)] is an ORNL-developed, one-dimensional, single-phase, finite-difference model for simulating forced and free convection transients in the THORS loop.

- 2678 (ORNL/TM--6782) INFLUENCE OF MELTING PRACTICE AND HEAT TREATMENT ON THE CONTINUOUS CYCLING FATIGUE AND SUBCRITICAL CRACK GROWTH PROPERTIES OF 2 1/4 CR--1 MO STEEL. Strizak, J.P.; Brinkman, C.R. (Oak Ridge National Lab., TN (USA)). May 1979. Contract W-7405-ENG-26. 33p. AT.

Interim results are reported for continuous cycling strain-controlled fatigue and fatigue crack growth tests of 2 1/4 CR--1 Mo steel. Heat-to-heat variations due to different melting practices and postweld heat treatment of annealed and isothermally annealed material were considered. Apparently, continuous cycle fatigue behavior at 427 and 538°C did not differ among the heats manufactured by air-melt, vacuum-arc remelt, and electro-slag remelt practices. Furthermore, a possible postweld heat treatment of 40 h at 726°C on initially annealed or isothermally annealed material did not influence continuous cycle fatigue life. Interim fatigue crack growth rate results at 510°C extended data to higher stress intensity levels. Fatigue crack growth rate data for isothermally annealed air-melt material obtained at frequencies of 0.067 and 0.67 Hz over a stress intensity range of 30 to 60 MPa $\sqrt{m}$ agreed well with previous results not exceeding 40 MPa $\sqrt{m}$. Cyclic life tests using both hourglass-shaped and uniform-gage specimens were compared. The 5.08-mm-diam hourglass-shaped specimen had a lower fatigue life than the 6.35-mm specimen. The discrepancy in the results for these two hourglass-shaped specimens decreases with decreasing total axial strain range. Furthermore, the uniform-gage specimen had a lower fatigue life than the 6.35-mm-diam hourglass-shaped specimen.

- 2679 (ORNL/TM--6792) SODIUM BOILING TESTS IN A 19-PIN INTERNALLY GUARD HEATED SIMULATED LMFBR FUEL ASSEMBLY WITH A SIX-CHANNEL INTERNAL BLOCKAGE: RECORD OF EXPERIMENTAL DATA FOR THORS BUNDLE 3C. Morris, R.H.; Nelson, W.R.;

Clapp, N.E.; Hanus, N.; Fontana, M.H.; Gnadt, P.A.; Montgomery, B.H.; Thornton, R.H.; Wantland, J.L. (Oak Ridge National Lab., TN (USA)). May 1979. Contract W-7405-ENG-26. 376p. AT.

A record is presented of experimental data obtained during boiling propagation tests with bundle 3C in the Liquid-Metal Fast Breeder Reactor (LMFBR) Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility. The objective of these tests was to verify that boiling in the wake of a local blockage will not propagate axially or radially. Bundle 3C consisted of 19 electrically heated pins [fuel pin simulators (FPSs)], having the same configuration and heat flux capability as Fast Flux Test Facility (FFTF) and Clinch River Breeder Reactor (CRBR) fuel, surrounded by 12 heated edge pins installed in a stainless steel duct. A 6.4-mm (0.25-in.) stainless steel plate attached to the central pin blocked the six central channels 381 mm (15.0 in.) downstream from the start of the heated section. The test bundle and the loop were designed to simulate prototypic boundary pressure conditions and downstream heat capacity characteristics.

- 2680 (ORNL/TM--6851) BREEDER REACTOR SAFETY AND CORE SYSTEMS PROGRAMS PROGRESS REPORT FOR OCTOBER--DECEMBER 1978. Fontana, M.H.; Wantland, J.L. (Oak Ridge National Lab., TN (USA)). Aug 1979. Contract W-7405-ENG-26. 111p. AT.

This report covers progress made during October--December 1978 by the Breeder Reactor Safety and Core Systems Programs at Oak Ridge National Laboratory (ORNL). Topics discussed are program coordination, thermal-hydraulic out-of-reactor safety (THORS) analysis and data evaluation, THORS Facility Operation Sodium Boiling Test (SBT) Facility, analytical models for energetics accommodation, environmental assessment of alternate Fast Breeder Reactor (FBR) fuels, model evaluation of breeder reactor radioactivity releases, Nuclear Safety Information Center (NSIC), central computerized data base for breeder reactor safety codes, and breeder reliability data analysis center.

- 2681 (ORNL/TM--6862) COMPARISON OF COMPUTATIONAL RESULTS OF THE SABRE LMFBR PIN BUNDLE BLOCKAGE CODE WITH DATA FROM WELL-INSTRUMENTED OUT-OF-PILE TEST BUNDLES (THORS BUNDLES 3A AND 5A). Dearing, J.F. (Oak Ridge National Lab., TN (USA)). Sep 1979. Contract W-7405-ENG-26. 41p. AT.

The Subchannel Analysis of Blockages in Reactor Elements (SABRE) computer code, developed by the United Kingdom Atomic Energy Authority, is currently the only practical tool available for performing detailed analyses of velocity and temperature fields in the recirculating flow regions downstream of blockages in liquid-metal fast breeder reactor (LMFBR) pin bundles. SABRE is a subchannel analysis code; that is, it accurately represents the complex geometry of nuclear fuel pins arranged on a triangular lattice. The results of SABRE computational models are compared here with temperature data from two out-of-pile 19-pin test bundles from the Thermal-Hydraulic Out-of-Reactor Safety (THORS) Facility at Oak Ridge National Laboratory. One of these bundles has a small central flow blockage (bundle 3A), while the other has a large edge blockage (bundle 5A). Values that give best agreement with experiment for the empirical thermal mixing correlation factor, FMIX, in SABRE are suggested. These values of FMIX are Reynolds-number dependent, however, indicating that the coded turbulent mixing correlation is not appropriate for wire-wrap

pin bundles.

- 2682 (ORNL/TM--6900) SENSITIVITY THEORY APPLIED TO A TRANSIENT THERMAL-HYDRAULICS PROBLEM. Weber, C.F.; Oblow, E.M. (Oak Ridge National Lab., TN (USA)). Oct 1979. Contract W-7405-ENG-26. 112p. Dep. NTIS, PC A06/MF A01.

A new method for sensitivity analysis of transient nonlinear problems is developed and applied to a reactor thermal-hydraulics problem. The method resembles the differential sensitivity methods currently used in the linear problems of reactor physics, but it is applicable to nonlinear systems as well. The equations governing heat transfer and fluid flow in a fuel pin and surrounding coolant are given and used to derive a second set of equations (commonly known as the adjoint equations) used in the sensitivity analysis. Both systems contain one second-order parabolic and one first-order hyperbolic partial differential equation. Difference equations are derived to approximate both systems and the convergence properties of these discrete systems are evaluated, yielding a useful analysis of the numerical solution. The solution functions are used to derive sensitivity coefficients for any desired integral response. These sensitivity coefficients are used in a first-order perturbation theory to predict changes in a response resulting from changes in parameter values. The results of a test problem are shown, verifying that this procedure is indeed useful for a wide variety of sensitivity calculations.

- 2683 (ORNL/TM--6947) ORNL BREEDER REACTOR SAFETY: LOA 2, 3, 4, AND 5. QUARTERLY TECHNICAL PROGRESS REPORT, JANUARY-MARCH 1979. Fontana, M.H. (Oak Ridge National Lab., TN (USA)). Sep 1979. Contract W-7405-ENG-26. 146p. AT.

Progress is summarized for the following lines-of-assurance (LOA) tasks: thermal-hydraulic out-of-reactor safety program, analytical models for energetics accommodation, environmental assessment of alternate FBR fuels, model evaluation of breeder reactor radioactivity releases, Nuclear Safety Information Center, breeder reactor reliability data analysis center, and central data base for breeder reactor safety codes.

- 2684 (ORNL/TM--6996) PRECONCEPTUAL STUDY OF PROLIFERATION-RESISTANT HETEROGENEOUS SPHERE-PAC-FUELED LMFBR CORES. Selby, D.L. (Oak Ridge National Lab., TN (USA)). Sep 1979. Contract W-7405-ENG-26. 206p. AT.

In October, 1977, a multi-institutional Proliferation-Resistant Preconceptual Core Design Study (PRPCDS) was initiated which extended to September, 1978. The objective of the study was to establish the proliferation resistance and performance potential of a wide range of sodium-cooled fast breeder reactor core designs. Each participating organization was responsible for publishing the results from its part of the study, and this report describes ORNL's work on the Vibro-Pac fuel design concept. Special characteristics of Vibro-Pac fuel (low density, etc.) were considered in terms of effects on Doppler coefficient, fuel time constant, breeding performance, etc., in a heterogeneous core configuration. A comparison was made with a reference pellet pin design. No significant differences between Vibro-Pac and high density pellet pins were identified.

- 2685 (ORNL/TM--6998) ELEVATED-TEMPERATURE DEFORMATION AND FAILURE TEST OF A TYPE 304 STAINLESS STEEL CAPPED CYLINDRICAL SHELL: SPECIMEN CC-2. Clinard, J.A.; Richardson, M.; Battiste, R.L.; Sartory, W.K. (Oak Ridge National Lab., TN (USA)). Oct 1979. Contract W-7405-ENG-26. 57p. AT.

Results are presented from a high-temperature deformation and failure test of a type 304 stainless steel capped cylindrical shell, specimen CC-2. The specimen was subjected to time-varying pressure and temperature which produced elastic-plastic-creep response. The test was terminated when failure occurred by opening of a creep crack which propagated circumferentially in the junction region between the cap and the cylinder. The test, which was performed to provide for assessment of the accuracy and validity of structural analysis methods and related computer programs, was conducted as part of the ORNL High-Temperature Structural Design (HTSD) Program effort related to development and verification of a design methodology applicable to liquid-metal fast breeder reactor (LMFBR) system components. The specimen was fabricated using a well-characterized heat (heat 9T2796) of type 304 stainless steel. Capacitive-type strain gages provided reliable strain-time response over the specimen life of 3300 hr. Complete test results are provided on microfiches. Analysis activities are discussed and computed results are compared to measured specimen response. Inelastic analysis predictions of the deformation phase were in reasonable agreement with strain-gage data. Failure predictions, based on both elastic and inelastic calculations, were conservative, although the degree of conservatism of inelastic failure predictions for this test could not be exactly determined since computational difficulties prevented complete solution of the large-deformation failure phase.

- 2686 (ORNL/TM--7086) SPQR: A MONTE CARLO REACTOR KINETICS CODE. Cramer, S.N.; Dodds, H.L. (Oak Ridge National Lab., TN (USA)). Feb 1980. Contract W-7405-ENG-26. 41p. Dep. NTIS, PC A03/MF A01.

The SPQR Monte Carlo code has been developed to analyze fast reactor core accident problems where conventional methods are considered inadequate. The code is based on the adiabatic approximation of the quasi-static method. This initial version contains no automatic material motion or feedback. An existing Monte Carlo code is used to calculate the shape functions and the integral quantities needed in the kinetics module. Several sample problems have been devised and analyzed. Due to the large statistical uncertainty associated with the calculation of reactivity in accident simulations, the results, especially at later times, differ greatly from deterministic methods. It was also found that in large uncoupled systems, the Monte Carlo method has difficulty in handling asymmetric perturbations.

- 2687 (ORNL/TM--7181) COVERX SERVICE MODULE OF THE FORSS SYSTEM. Drischler, J.D. (Oak Ridge National Lab., TN (USA)). Apr 1980. Contract W-7405-ENG-26. 42p. (ENDF--291). Dep. NTIS, PC A03/MF A01.

The COVERX Service Module includes seven execution paths to aid in understanding and using multigroup cross-section covariance matrices contained in the standard interface file COVERX. The execution paths provide the following operations on COVERX file(s): list

the contents of a COVERX file; allow adding new multigroup cross-section covariance matrices to an existing COVERX file; allow deletion of multigroup covariance matrices from an existing COVERX file; merge two COVERX files and creates a new file; change the mode of a file from unformatted to formatted and conversely; allow modification of the records contained in a COVERX file; and selectively edits or copies a file.

- 2688 (ORD--5295-1) BENEFIT-COST ANALYSIS FOR DYNAMIC TESTING IN THE FAST FLUX TEST FACILITY. Huynh, Q.A.; Kerlin, T.W. (Tennessee Univ., Knoxville (USA). Dept. of Nuclear Engineering). Jun 1978. Contract EN-77-S-05-5295. 341p. Dep. NTIS, PC A15/MF A01.

Thesis.

Dynamic tests for the Fast Flux Test Facility (F.F.T.F.) were studied to determine their potential benefits and their costs. This required the development of a mathematical model for the plant. Models were developed with several different treatments of the heat transfer and the reactivity feedback effects to determine the model complexity required to represent the system adequately. The model was used to assess the benefits of dynamic testing. Specifically, sensitivity studies were performed to evaluate the identifiability of important performance-related plant parameters (such as reactivity feedback coefficients and heat transfer coefficients). Also, a test was designed that uses only existing in-plant hardware and instrumentation. It was concluded that the potential benefits of dynamic testing in the F.F.T.F. are high and the costs are low.

- 2689 (PB--254737, pp 324-359) FEASIBILITY STUDIES OF ACTINIDE RECYCLE IN LMFBRs AS A WASTE MANAGEMENT ALTERNATIVE. Beaman, S.L.; Aitken, E.A. 1976.

Proceedings of the conference on the management of radioactive waste: waste partitioning as an alternative.

Actinide recycle in LMFBRs offers an attractive alternative on long-term storage of the actinides. The concept will not significantly affect the performance of the LMFBR, but will affect other parts of the nuclear fuel cycle. Assuming that hands-on maintenance will be allowed for Pu-recycle fuel fabrication facilities, the transplutonium actinides should be kept separate from the PuO_2 - UO_2 fuel. Thus, the "reference" recycle scheme should be defined as a scheme in which the actinides are recycled in target assemblies. The target assemblies should be reprocessed either in batches separate from spent-fuel batches or in a separate, relatively small, special purpose reprocessing plant. The target assemblies should be fabricated in a special purpose, remotely maintained facility.

- 2690 (PB--263917) REPORT OF THE AD HOC PANEL ON THE USE OF ACCELERATORS TO STUDY IRRADIATION EFFECTS. Reed, C.K. (National Academy of Sciences - National Research Council, Washington, DC (USA). Committee on Nuclear Science). 1974. Contract NSF-C310. 64p. NTIS PC A04/MF A01.

A panel was assembled under the auspices of the National Academy of Sciences, Committee on Nuclear Sciences to review the ways nuclear physicists might apply their skills and instrumentation to problems in material sciences. The panel looked into problems related to the effects of radiation on dimensional stability, mechanical properties, structural and compositional changes and

surface effects. The role the accelerator might play in these studies and the effects of specific particle irradiation is presented in some detail.

- 2691 (PB--273304) CONDENSATION OF FUEL ONTO THE ABOVE-CORE STRUCTURE DURING AN LMFBR CORE-DISRUPTIVE ACCIDENT. Garner, P.L.; Erdman, C.A.; Reynolds, A.B. (Virginia Univ., Charlottesville (USA). Dept. of Nuclear Engineering). Oct 1977. Contract AT(49-24)-0316. 157p. (UVA--529106/NE77/101). NTIS PC A08/MF A01.

Condensation of a pure, saturated vapor onto a vertical, melting substrate is analyzed for both one- and two-material situations. The condensate and melt regions are treated as laminar boundary layers in the steady-state analyses. The convective momentum and energy transport terms in the conservation equations and the shear stress at the condensate/vapor interface are found to be non-negligible; neglect to these terms results in overestimating the phase-change rates by a factors of 2 to 3. The phase-change rates are reduced if the vapor is moving upward at low speed rather than stagnant.

- 2692 (PB--274091) THE CLINCH RIVER BREEDER REACTOR PLANT: AN ANALYSIS OF THE IMPACTS OF ITS IN-MIGRANT CONSTRUCTION WORKERS ON LOCAL PUBLIC SERVICES. FINAL REPORT. Braid, R.B. Jr.; Kyles, S.D. (East Tennessee Energy Projects Coordinating Committee, Clinton (USA)). May 1977. Contract ARC-TQ-10/TN-5105-76-I-302-0401. 328p. NTIS PC A15/MF A01.

The socioeconomic impact study identifies certain impacts which are projected to occur to local public services in each of 14 Tennessee communities in the Oak Ridge-Knoxville area during the construction of the Clinch River Breeder Reactor Plant. Various in-migration scenarios are utilized, and detailed qualitative and quantitative analyses of each public service are undertaken. Per capita in-migrant cost-revenue impacts are calculated for each community in each in-migration scenario.

- 2693 (PB--280695) A LOOK AT ALTERNATIVE CORE DISRUPTION ACCIDENTS IN LMFBR'S. PART II. A GENERAL EVALUATION APPROACH TO RISK-BENEFIT FOR LARGE TECHNOLOGICAL SYSTEMS AND ITS APPLICATION TO NUCLEAR POWER. Badham, V.; McKone, T.K.; Steele, W.; Chan, C.K. (California Univ., Los Angeles (USA). Dept. of Chemical, Nuclear, and Thermal Engineering). Dec 1977. 59p. (UCLA-ENG--7776). NTIS PC A04/MF A01.

The report investigates the effects of fuel behavior and control rod motion of an LMFBR core in a postulated accident scenario, in which the heatup of the fuel is caused mainly by decay heat. The sequence of this class of accidents is usually characterized by the loss of sodium, cladding, and can wall before major fuel melting occurs. There are two possible paths for the core to gain reactivity: either as a result of fuel slump or by loss of control rod material from the active core region. A parametric approach is adopted as a first order estimation for the possibility and manner of achieving recriticality. A major factor in preventing fuel slump before fuel melting is fuel swelling at elevated temperature. A model is developed to estimate the intragranular fission gas behavior and the subsequent fuel swelling after the loss of the subassembly wall. The heat radiated from the fuel pin would cause the melting of the structural material of the control rod. The loss of the control rod

material from the active core would cause a ramp insertion of reactivity. Because of the complex nature of core geometry, the core transient behavior is studied in terms of the initial temperature, ramp rates, and other reactor parameters.

- 2694 (PB--283030) A QUALIFICATION TESTING PROGRAM PLAN FOR SIMMER; A COMPUTER CODE FOR LMFBR DISRUPTED CORE ANALYSIS. RESEARCH PROGRAM PLANNING REPORT. Basdekas, D.L.; Silberberg, M.; Curtis, R.T.; Kelber, C.N. (Nuclear Regulatory Commission, Washington, DC (USA)). Sep 1978. 52p. NTIS PC A04/MF A01.

The SIMMER Code, developed at Los Alamos Scientific Laboratory, analyzes the various phases of a CDA (Core disassembly accident) in an LMFBR. The qualification testing program, comprised of two stages, is discussed. The first stage is developmental verification, which involves interaction with code development. The second stage is independent qualification testing and it involves an independent audit of the first stage and confirmatory predictions of large experiments without a priori adjustments of the Code. An experimental and analytical approach is described intended to accomplish this task.

- 2695 (PB--283231) MULTIPHASE TRANSIENTS WITH COOLANT AND CORE MATERIALS IN LMFBR CORE DISRUPTIVE ACCIDENT ENERGETICS EVALUATIONS. TECHNICAL REPORT. Theofanous, T.G.; Beecher, J.; Miao, C.; Marrotte, G.; Patel, P. (Purdue Univ., Lafayette, IN (USA)). School of Nuclear Engineering. Jul 1978. Contract NRC-03-76-181. 302p. NTIS PC A14/MF A01.

An overview of core disruptive accidents in LMFBRs and an assessment of their potential for energetic consequences are presented. Areas in which current state-of-the-art is found inadequate are identified. In particular, the early post disruptive period (called herein Transition to the Transition Phase) is identified as a key area having received inadequate attention to date. Further the idea of mild dispersal from a fully developed Transition Phase at decay heat power levels does not appear justified. However, considerable questions exist as to the attainability of such full core pool under accident conditions. Specific recommendations for research work in relation to licensing needs are presented.

- 2696 (PB--284227) PROCEEDINGS OF THE WORKSHOP ON INSTITUTIONAL ALTERNATIVES FOR LMFBR DEVELOPMENT AND COMMERCIALIZATION HELD AT THE MITRE CORPORATION IN MCLEAN, VIRGINIA ON SEPTEMBER 8 AND 9, 1976. FINAL REPORT. (Mitre Corp., McLean, VA (USA)). METREK Div.). Jul 1977. 513p. NTIS PC A22/MF A01.

The workshop assembled representatives of utilities, nuclear vendors, and government and research organizations to consider institutional issues of the development and commercialization of liquid metal fast breeder reactors (LMFBR). The document contains transcripts of the presentations as well as discussions in several topical areas. The Energy Research and Development Administration's LMFBR program was outlined. Several studies of the development and commercialization of light water reactors were presented as background and a basis of discussion. Nuclear technology development and commercialization in foreign countries were outlined in several presentations, suggesting alternatives for development of LMFBR technologies in this country. The critical

stage appears to be the transition between R and D and commercialization, i.e., the first thirty commercial reactors, and the appropriate government role at this stage needs to be defined.

- 2697 (PB--285544) SAFETY, SAFEGUARDS, AND ENVIRONMENTAL EFFECTS OF LIQUID METAL FAST BREEDER REACTORS IN RELATION TO LICENSE ABILITY AND PUBLIC ACCEPTANCE. FINAL REPORT. O'Leary, J.F. (National Science Foundation, Washington, DC (USA)). Div. of Policy Research and Analysis. 3 Sep 1974. 28p. NTIS PC A03/MF A01.

Public concerns with the Liquid Metal Fast Breeder Reactor (LMFBR) have emerged in three areas: the relative environmental impact of this energy source, the potential for reactor accidents, and the risk of plutonium diversion. Compared with other central station electricity generation options, the LMFBR is environmentally more benign. More reliable engineered defenses against results of a voiding accident can possibly be developed, but the added protection of remote siting should be used. The safeguards issue is the principal obstacle to public acceptance, and the best attempts at quantification have shown the risk to public health and safety from diversion significantly greater than risk of reactor accident. Siting policy options offer the best means of reducing this risk by reducing the physical transport of plutonium in various stages of the fuel cycle. Integrated fuel cycle facilities, integrated LMFBR facilities, and integrated nuclear generating facilities are three such options for the integration concept that may have other attractions as well.

- 2698 (PB--285567) ENVIRONMENTAL, SAFETY, AND PUBLIC ACCEPTANCE ASPECTS OF LMFBRs. MEMO REPORT. Moss, L.I. (National Science Foundation, Washington, DC (USA)). Office of Energy Research and Development Policy. 9 Sep 1974. 50p. NTIS PC A03/MF A01.

The liquid metal fast breeder reactor (LMFBR) offers a number of important benefits, but should be evaluated in the context of a more rational energy policy than the present policy. Elimination of subsidies and internalization of social and environmental costs, reordering RandD priorities to reflect environmental acceptability, reassessment of the growth rate of US electrical energy demand, and availability of domestic high-grade uranium ores are all necessary energy policy reforms that will affect the supply role of LMFBRs. LMFBRs have some environmental, health and safety advantages over light water reactors, but also some disadvantages. A program to capture the advantages and minimize the hazards of LMFBRs should include development of integrated nuclear centers, safety analysis and experiments to quantify hazards, assessing underground siting, and developing the capability to separate and recycle actinides through LMFBRs.

- 2699 (PB--286865) EXPERIMENTAL INVESTIGATION OF AEROSOL BEHAVIOR IN A POST-LMFBR ACCIDENT REACTOR CONTAINMENT ATMOSPHERE. ANNUAL REPORT SEP 1977--JAN 1978. Wegrzyn, J.; Shaw, D.T. (State Univ. of New York, Buffalo (USA)). Dept. of Electrical Engineering. Jun 1978. Contract AT(49-24)-0359. 35p. NTIS PC A03/MF A01.

Under the support of the Nuclear Regulatory Commission, work has been done at SUNY=Buffalo for the year 1977 on the experimental measurements of the aerodynamic properties of a sodium aerosol. With this format essentially

three separate experiments were performed; (1) the evaluation of the centrifuge as a in-situ measuring device, (2) the classification of the sodium aerosol, and (3) the shape factor measurements on an exploding wire (E.W.) aerosol.

2700 (PB--287481) NATURAL CONVECTION HEAT TRANSFER IN BEDS OF INDUCTIVELY HEATED PARTICLES. TOPICAL REPORT. Rhee, S.J.; Dhir, V.K. (California Univ., Los Angeles (USA)). Aug 1978. Contract NRC-04-76-246. 99p. NTIS PC A05/MF A01.

The estimation of heat-removal-capability from core debris that may be formed as a result of a hypothetical fuel disruptive accident in a LMFBR is of interest in the nuclear reactor safety analysis. After the accident, the fuel debris in the form of particles may settle on various horizontal surfaces, such as thermal baffles, and there may exist an overlying sodium layer above the core debris bed. The decay heat from the fuel debris bed will be removed either by conduction or convection across the superposed layer before the initiation of ebullition. Experimental observations of the onset of convective motion in such beds of inductively heated particles have been made. The data for the heat transfer coefficient have been obtained in water-cooled beds of 304 stainless steel particles. The observations show that increasing the depth of the fluid layer over the bed tends to lower the critical internal Rayleigh number at which the onset of convection occurs. For overlying fluid-layer-depth-to-particulate-bed-depth ratios of 1 of greater, natural convection has been observed to begin at RI approximately equal to 12. However, for no fluid layer at the top of the bed, natural convection has been observed to occur at R sub I approximately equal to 45, which is about the same as observed in earlier studies made with Joule heating and no fluid layer at the top of the bed.

2701 (PB--293190) BUBBLE BEHAVIOR IN LMFBR CORE DISRUPTIVE ACCIDENTS. ANNUAL REPORT, 1 OCTOBER 1977-30 SEPTEMBER 1978. Erdman, C.A.; Reynolds, A.B.; Bradley, D.R.; Chen, K.; Pilch, M. (Virginia Univ., Charlottesville (USA)). Dept. of Nuclear Engineering and Engineering Physics. Mar 1979. Contract AT(49-24)-0316. 124p. NTIS, PC A06/MF A01.

Analytical research performed at the University of Virginia during FY78 for the Reactor Safety Research (RSR) Division of the U.S. Nuclear Regulatory Commission is reported. The research is part of the Aerosol Release and Transport program of RSR. Three major areas were investigated: (1) particle production by hydrodynamic fragmentation, (2) particle production by homogeneous nucleation, and (3) bubble dynamics and heat transfer. Development of a Monte Carlo computer code for the analysis of particle-size distributions from vapor-liquid hydrodynamic fragmentation was initiated. The source of homogeneous nucleation in the ORNL CDV test was postulated to be expansion behind a spherical shock wave and modeling of this phenomenon was initiated. Analysis of the axial motion of bubble in the French EXCOBULLE tests was completed with the use of the EXCOBUL code in preparation for its use in assisting with the analysis of the ORNL under-water and under-sodium CDV tests.

(California Univ., Los Angeles (USA)). School of Engineering and Applied Science). Mar 1979. Contract NRC-03-77-001. 82p. NTIS, PC AA05/MF A01.

This report presents a review of post-accident heat removal problems and views upon areas that include debris bed phenomena, molten pool growth, gas film boundaries and crusts, and the influence of basemat conduction upon pool formation and evolution. It introduces plausible circumstances for the formation of non-coolable debris beds, and so leads to crude estimates of the initial dimensions of core debris distributions for pool growth problems. The GROWS code is reviewed and examples given using initial conditions derived from those estimates. Gas film boundaries with crusting and freezing phenomena are qualitatively discussed as potentially significant factors in defining heat split and pool evolution. Research on density gradient mixing in dense pools overlying light sacrificial materials is reviewed. The conductive transport of heat in the solid basemat surrounding the pool is discussed as a factor that might influence the initial spreading of a molten mass upon a basemat, may influence the shape of a pool when strong top surface cooling is present, and is a dominant factor in defining the cessation of penetration.

2703 (PB--294243) LMFBR FUEL ANALYSIS. TASK C: RELIABILITY. FINAL REPORT, 1 OCTOBER 1977-15 NOVEMBER 1978. Cave, L.; Kastenber, W.E. (California Univ., Los Angeles (USA)). School of Engineering and Applied Science). Mar 1979. Contract NRC-03-77-001. 30p. NTIS, PC AA03/MF A01.

This report consists of consultative support on a variety of topics, the main items being (1) evaluation of certain aspects of the Fast Flux Test Facility design; (2) examination of means of alleviating the effects of systems interaction; and (3) evaluation of the suitability for licensing in the United States of alternative reactor systems, and of alternative fuel cycles for existing systems, which might assist in the reduction of proliferation risks. In the case of one particular topic, the problem of establishing an adequate verification program for natural circulation in the Fast Flux Test Facility, a probabilistic approach was developed which may be of more general interest in relation to other fast reactors.

2704 (PB--295077) EXPERIMENTAL INVESTIGATION OF AEROSOL BEHAVIOR IN A POST-LMFBR ACCIDENT REACTOR CONTAINMENT ATMOSPHERE. ANNUAL REPORT, 1 OCTOBER 1977-30 SEPTEMBER 1978. Wegrzyn, J.; Shaw, D.T. (State Univ. of New York, Buffalo (USA)). Lab. for Power and Environmental Studies). Apr 1979. Contract NRC-04-76-359. 52p. NTIS, PC A04/MF A01.

For large-scale testing of sodium-fire aerosols, no off-the-shelf instrument is available for accurate measurement of the particle size distribution function in the diameter region from 0.5 - 30 μ m. In the long range, cascading impactors are considered to be most suitable for in-situ measurement of sodium-fire aerosols because of simplicity of operation and maintenance. The calibration of cascading impactors can be best carried out by a combined use of the centrifuge (for particles with diameter range from 0.5 - 5 μ m) and by direct observation with a microscope (for particles ranging from 5 - 30 μ m). Three distinctly different characteristics are found. To avoid the complex problem of controlling the chemical composition of the sodium-fire aerosol during and after the particle generation, it is

2702 (PB--294102) LMFBR FUEL ANALYSIS. TASK B: POST-ACCIDENT HEAT REMOVAL. FINAL TECHNICAL REPORT 1 OCTOBER 1977-15 NOVEMBER 1978. Catton, I.; Dhir, V.K.; Brinsfield, W.A.; McDonough, J.M.; Ostrander, N.C.

best to analyze the aerosol by first classifying the particles according to their size - with either a centrifuge or a cascading impactor - and then to determine the chemical composition of the particles in each size range by electron diffraction technique. Very limited information is available at the present time for the particle size distribution measurement in the diameter range 5 - 30 m.

- 2705 (PEL--250) DEVELOPMENT OF ACCURATE DATA FOR THE DESIGN OF FAST REACTORS. Rossouw, P.A. (Pretoria Univ. (South Africa). Faculty of Mathematics and Natural Sciences; Atomic Energy Board, Pelindaba, Pretoria (South Africa)). Apr 1976. 99p. (In Afrikaans). Dep. NTIS (US Sales Only), PC A05/MF A01.

The proposed use of nuclear power in the generation of electricity in South Africa with the use of fast reactors in the country's nuclear program requires a method for fast reactor evaluation. The availability of accurate neutron data and neutronics computation techniques for fast reactors is required for such an evaluation. The reactor physics and reactor parameters of importance in the evaluation of fast reactors are discussed, and computer programs for the computation of reactor spectra and reactor parameters from differential nuclear data are presented. In endeavouring to increase the accuracy in fast reactor design, two methods for the improvement of differential nuclear data were developed and are discussed in detail. The computer programs which were developed for this purpose are also given. The neutron data of the most important fissionable and breeding nuclei (U^{235} x U^{238} x Pu^{239} and Pu^{240}) are adjusted using both methods, and the improved neutron data are tested by computation with an advanced neutronics computer program. The improved and original neutron data are compared, and the use of the improved data in fast reactor design is discussed.

- 2706 (PNC-J--225-78-04) RESEARCH REPORT ON DESIGN ALLOWABLE VALUES OF STRUCTURAL MATERIALS FOR LMFBR. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Nov 1978. 216p. (In Japanese). NTIS (US Sales Only), PC A11/MF A01.

The present report is composed of the following two main parts. i) review and re-evaluation on test results by FCI Sub-committee studies, performed from 1973 to 1976, ii) review on procedures for determining design allowable values of structural materials for LMFBR components. Re-evaluation works have been made on monotonic tensile properties at elevated temperatures, creep and creep rupture properties, creep-fatigue properties (strain rate and tensile strain hold time effects on strain fatigue properties at elevated temperatures) of Types 316 and 304 stainless steel and 2 1/4Cr-1Mo steel (base and weld metals) produced in Japan. In the first half of the present report, creep-fatigue test results obtained by FCI Sub-committee studies are subjected to re-evaluation by the present P-FCI Sub-committee. Reviews have been made on testing methods on FCI's-creep-fatigue experiments with other test data of the test materials; high temperature monotonic tensile data, creep and creep rupture data, and origin of the test materials. The data of FCI studies are compared with other reference data obtained by several Japanese laboratories. In the latter half of the present report, procedures including ASME's are reviewed for setting design allowable values for LMFBR components on the basis of high temperature strength properties obtained with materials produced in

Japan. A creep rupture data of Japanese steels are issued and examined to make proposal for a design allowable stress of $S_{sub}(t)$ through parameter survey.

- 2707 (PNC-N--251-76-12(Vol.1), pp 25-32) REVIEW AND PERSPECTIVE OF FCI RELATED ACTIVITIES IN JAPAN. Kawaguchi, O. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

LMFBRs are the most promising reactors among several nuclear options, and the safety must be established for the future commercialization. One of the most crucial issues in the LMFBR safety is the molten fuel-coolant interaction which takes place in the various phases of hypothetical core disruptive accident. The objectives of this paper are to summarize the current considerations on the FCI applied to the design of the prototype FBR, Monju, in Japan, and to review the supporting R and D activities. The calculation with the codes SUGAR and DETECT on the initiation of the pin-level FCI, the comparison of transient overpower and loss of flow cases in the initiating phase of the whole core accident, the analysis of the mechanical disassembly phase with the upgraded version of VENUS, the treatment of the structural response with PISCES-2DL or two codes ASPRIN 2 and CAMEL, and the systematic approach with SIMMER to the post-accident heat removal are explained. In order to support the Monju design described above, several basic experiments have been in progress to elucidate the FCI aspects, such as liquid-liquid contact and thermal explosion, the effect of the boundary conditions surrounding FCI zones on vapor explosion, the efficiency of converting stored thermal energy to mechanical work, the fragmentation of molten metal, and the incoherence effects in FCI. The out-of-pile and in-pile experiments on the UO_2 -Na system have also been carried out since 1969.

- 2708 (PNC-N--251-76-12(Vol.1), pp 285-305) SIMULATION EXPERIMENT AND ANALYSIS ON THE EFFECTS OF IN-COHERENCE IN FUEL COOLANT INTERACTION. Kondo, S.; Togo, Y.; Iwamura, T. (Tokyo Univ. (Japan). Faculty of Engineering). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

Experimental and analytical studies were conducted to investigate effects of incoherence (space time behavior of molten fuel) on molten fuel coolant interaction. In experiments, a 2 mm diameter molten tin jet was injected upward into the water in a slender tank. The results were analyzed based on the pressure records and high speed photographs. The pressure records indicated that there were two types of interaction between molten jet and water, intermittent explosion mode and continuous one. The explosion mode appeared when the temperature of molten tin was above 350°C or so and that of water was below 70°C or so. The high speed photograph indicated that an establishment of a stable jet column was necessary for an explosive interaction and that a bubble like region grew and collapsed at the root of the jet in accordance with the generation of pressure pulse. It was found that the mass of metal which contributed to the vapor explosion was only a small part of the injected metal in the case of jet injection type contact mode and this was the reason why the gross thermal to mechanical energy conversion ratio was around 0.03% in this type of contact mode, though this ratio was around 2% if only the part of record around the

pressure pulse was taken into consideration. In the analysis part, a multi-channel FCI model was developed to evaluate the spatial incoherence effect on pressure at subassembly exit. The calculated pressure trace indicated that the spatial incoherence has considerable effects for an evaluation of structure response under FCI pressure loading.

- 2709 (PNC-N--251-76-12(Vol.1), pp 97-127)
FUEL-COOLANT INTERACTIONS. SOME BASIC STUDIES AT THE UKAEA CULHAM LABORATORY. Reynolds, J.A.; Dullforce, T.A.; Peckover, R.S.; Vaughan, G.J. (UKAEA Research Group, Abingdon. Culham Lab.). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

In a hypothetical fault sequence important effects of fuel-coolant interactions include voiding and dispersion of core debris as well as the pressure damage usually discussed. The development of the fuel-coolant interaction probably depends on any pre-mixing Weber break-up that may occur, and is therefore a function of the way the fuel and coolant come together. Four contact modes are identified: jetting, shock tube, drops and static, and Culham's experiments have been mainly concerned with simulating the falling drop mode by using molten tin in water. It was observed that the fuel-coolant interaction is a short series of violent coolant oscillations centred at a localized position on the drop, generating a spray of submillimeter sized debris. The interaction started spontaneously at a specific time after the drop first contacted the water. There was a definite limited fuel-coolant interaction zone on a plot of initial coolant temperature versus initial fuel temperature outside which interactions never occurred. The interaction time was a function of the initial temperatures. Theoretical scaling formulae are given which describe the fuel-coolant interaction zone and dwell time. Bounds of fuel and coolant temperature below which fuel-coolant interactions do not occur are explained by freezing. Upper bounds of fuel and coolant temperatures above which there were no fuel-coolant interactions are interpreted in terms of heat transfer through vapour films of various thicknesses.

- 2710 (PNC-N--251-76-12(Vol.1), pp 381-442)
SOME EXPERIENCES AND INPILE EXPERIMENTAL PROGRAMME ON FUEL COOLANT INTERACTION IN NSRR. Saito, S.; Fujishiro, T.; Inabe, T.; Ishikawa, M. (Japan Atomic Energy Research Inst., Tokai, Ibaraki. Tokai Research Establishment). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

One of the main interests in the phase-I programme of the NSRR (Nuclear Safety Research Reactor) project is to investigate the degree of fuel dispersal into the coolant, molten fuel-coolant interaction and its mechanical effect in the reactor condition. The NSRR is a large-pulsed reactor, its maximum pulsing performance attained with 4.67 dollar reactivity insertion is energy release of 117 MW.sec, peak power of 21,100 MW at a minimum reactor period of 1.17 msec. The phase-I experiments started in October, 1975, and more than ten in-pile tests have been performed in water environment with ambient pressure and temperature condition so far. Among them fuel failure occurred in two cases and in 292 cal/g.UO₂ test, the test fuel was almost completely melted and broken into large five pieces and accompanying particles. Even in this case, however, detectable pressure generation was not observed. Several hundred tests are planned in the phase-I programme for water reactor fuels and fast reactor fuels in

water environment and in sodium environment, respectively. The out-of-pile experiment on fuel-coolant interaction has been also performed using several sorts of simulant metals mainly to find out the thermal conditions for fragmentation. Some interesting results on the thermal thresholds for occurrence of fragmentation were obtained. The analytical model for evaluation of fuel-coolant interaction has been developed. The compressibility of liquid sodium and fragmentation process under simple assumption are taken into account in the model. The effects of several sensitive parameters, for instance, mixing time, the presence of non-condensable gas in the interaction region, heat transfer coefficient between fuel particle and sodium, and so on, on the amounts of peak pressure and mechanical energy have been studied quantitatively.

- 2711 (PNC-N--251-76-12(Vol.1), pp 467-506)
FUEL COOLANT INTERACTION RESULTS IN THE FUEL PINS MELTING FACILITY(PMF). Uruwashi, S.; Hirabayashi, T.; Mizuta, H. (Power Reactor and Nuclear Fuel Development Corp., Tokai, Ibaraki (Japan). Tokai Works). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

The experimental work related to FCI at PNC has been concentrated to molten UO₂-dropping test. The primary objectives of the fuel pin-melting facility test are to obtain the detailed experimental results on the FCI under transient overpower and loss of flow conditions, to observe the movement of fuel before and after the failure of a pin by X-ray cinematography, to simulate the fuel behavior in the pin-pile test, and others. In the experimental apparatus, the UO₂ pellets clad in SUS 304 tubes are melted by direct electrical heating in a liquid sodium column, and the molten UO₂ coming out of the cladding tubes interacts with the surrounding sodium. The variation of electrical voltage and current, temperature, pressure and the movement of the sodium column are measured. The results of four capsule tests are presented in this paper, two for the open type test section and two for the closed type. Although the fuel motion and the FCI in the closed test section appeared to be different from those in the open test section, more experimental data are required to come to a conclusion. The best heating condition for UO₂ pellets in the FCI study with the pin-melting facility was able to be established. Further development will be continued for providing thermocouples in UO₂ pellets, increasing the stack length of pellets, visualizing the movement of fuel, and forwarding the loop test.

- 2712 (PNC-N--251-76-12(Vol.1)) PROCEEDINGS OF THE THIRD SPECIALIST MEETING ON SODIUM/FUEL INTERACTION IN FAST REACTORS. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). 1976. 513p. (CONF-760328--Pi). Dep. NTIS (US Sales Only), PC A22/MF A01.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

Each paper has been separately processed for the Energy Data Base.

- 2713 (PNC-N--251-76-12(Vol.1), pp 187-229)
SIMPLE AND CONSISTENT EQUATION OF STATE FOR SODIUM IN THE SINGLE PHASE AND TWO PHASE REGIONS. Breton, J.P. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides). 1976.

From 3. specialists' meeting on sodium fuel

interaction; Tokyo, Japan (Mar 1976).

An equation of state valid over an extended temperature and density range has been derived. Then, the following properties have been deduced: coefficient of thermal expansion, isothermal coefficient of bulk compressibility, thermal pressure coefficient, heat capacity at constant pressure, at constant volume, along the saturation curve for liquid, for vapor, heat of vaporization, speed of sound, and finally the Mollier diagram and the entropy diagram. All the obtained properties are thermodynamically consistent and satisfy the basic relations of thermodynamics for both single phase and two-phase regions. Experimental results were always used when available.

- 2714 (PNC-N--251-76-12(Vol.1), pp 231-248)
VAPOR EXPLOSION WITH SUBCOOLED FREON. Henry, R.E.; Fauske, H.E.; McUmber, L.M. (Argonne National Lab., Ill. (USA)). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

A set of experiments utilizing a well-wetted system of freon and mineral oil was performed to provide a more definite test of the interface temperature representation by subcooling cold liquid. These experiments were conducted in two different manners to investigate the relative importance of the interface temperature. The first was performed with freon initially at its normal boiling point and mineral oil at the temperature ranging from 80 to 220 deg C in 10 degree increments, and the second was carried out with the mineral oil temperature held constant at 190 deg C and decreasing freon temperature from -41 to -14C deg C in 20 degree decrements. The tests were carried out in the apparatus similar to that discussed in the previous report. The principal difference between the facilities is that in this case, the interaction vessel always contained freon, and hot oil was poured into the vessel. This change in contact state has little significance when freon-22 is saturated, but for subcooled freon, it provides better experimental resolution for the initial freon temperature. These experiments demonstrate that the interface temperature model not only allows to predict the explosive range of a given pair of liquids, but also enables to correlate the maximum explosive pressure which could be produced by a system which lies within the explosive limits.

- 2715 (PNC-N--251-76-12(Vol.1), pp 61-74)
IMPORTANCE OF SODIUM FUEL INTERACTION IN FAST REACTOR SAFETY EVALUATION. CEA POINT OF VIEW. Tanguy, P. (CEA, 75 - Paris (France)). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

The consequences of the interaction between molten metal (aluminum-uranium alloy) and water have long been a subject of concern for those in charge of reactor safety, following the accidents observed in BORAX, SL1, and SPERT 1D reactors. Similarly, the consequences of the possible contact between molten mixed uranium and plutonium oxides and sodium must be carefully studied, since such contact may occur in the accident conditions in sodium is explicitly mentioned in the recommendation made by a group of experts in France concerning the safety of fast neutron nuclear power stations. None of the in- or out of pile experiments carried out to date on UO_2 and sodium showed violent phenomena, except three out of pile tests by Armstrong. Recently, the work by Fauske on spontaneous nucleation and that by Board and Hall have contributed to the better understanding of the phenomena. In order to

define clearly the importance of the interaction phenomena, the accident within an assembly due to blockage and the overall accident concerning a core were examined. The violent interaction caused by the blockage in an assembly is unlikely. In case of the overall accident on the core of a fast reactor, the interaction is the consequence of the retention of mechanical energy and the recovery of fuel fragments in overall cooled configuration with local cooling problems.

- 2716 (PNC-N--251-76-12(Vol.1), pp 341-361)
EXPERIMENTAL WORK AT HARWELL ON THE INJECTION OF SODIUM INTO LIQUID STEEL. Asher, R.C.; Bradshaw, L.; Collett, R.; Davies, D. (UKAEA Research Group, Harwell. Atomic Energy Research Establishment). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

The equipment and experimental technique used to inject molten sodium beneath the surface of molten steel (and subsequently molten uranium dioxide) are described. The results of an exploratory experiment, in which 2 g of sodium at 380°C were injected into 54 g of steel at 1530°C, are outlined. A violent reaction occurred, apparently as a result of a vapour explosion, and a number of pulses were recorded by a force transducer. A preliminary examination of the steel debris is reported.

- 2717 (PNC-N--251-76-12(Vol.1), pp 443-465)
CURRENT STATUS OF EXPERIMENTAL AND THEORETICAL WORK ON SODIUM/FUEL INTERACTION(SFI) AT KARLSRUHE. OUT-OF-PILE EXPERIMENTS. Beutel, H. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Reaktorentwicklung); Bojarsky, E.; Caldarola, L.; Jacobs, H.; Reiser, H. 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

In April 1975, the operation of the sodium loop test facility for out-of-pile experiments started at the Institute of Materials and Solidstate Research. In this facility, it is possible to melt down single UO_2 rods and the small bundles of up to 7 rods in sodium flow by heating them electrically. The rods are subjected to power transient, starting from steady operating conditions. This gives rise to rod failure and to fuel-sodium interaction within several hundreds of msec. During the power excursion, pressure, flow, electrical voltage and current are measured. Prior to the power excursion, the conditions can be established in the cooling channels, which are comparable to those existing in the SNR fuel element. In one of the experiments, the cladding melted at the bottom of the fuel zone after the power excursion had been initiated, and liquid UO_2 was released into sodium flow. This produced very short pressure peaks in the test section. The major change of flow took place at the same time. The temperature profile inside the fuel rod was calculated with the computer program ELEKTRA. It must be concluded that the all interactions were very mild. The works in cooperation with the European Nuclear Research Center at Ispra have been carried out, and hot materials fractured in all tests, but no energetic interaction was observed.

- 2718 (PNC-N--251-76-12(Vol.1), pp 249-283)
RECENT ADVANCES IN UNDERSTANDING LARGE SCALE VAPOUR EXPLOSIONS. Board, S.J.; Hall, R.W. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

The whole process of efficient large scale

vapour explosions is considered in the following three stages; a) the setting up of a quasi-stable initial configuration, b) the triggering of this configuration, and c) the propagation of the explosion. Each stage is considered in turn, with the review of the relevant experimental information and theory to see to what extent the requirements for energetic explosions, and the physical processes that can satisfy these requirements, are understood. Particular attention is paid to an attractively simple criterion for explosivity, suggested by Fauske, that the contact temperature should exceed the temperature for spontaneous nucleation of the coolant, because on this criterion, sodium and uranium oxide in particular are not explosive.

- 2719 (PNC-N--251-76-12(Vol.1), pp 75-95)
EXPERIMENTAL STUDIES OF THERMAL INTERACTIONS AT AEE WINFRITH. Briggs, A.J. (UKAEA Reactor Group, Winfrith. Atomic Energy Establishment). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

The assessment of the consequences of very severe reactor accidents in which fuel melts involves the evaluation of the nature of any thermal interaction between the molten fuel or cladding and reactor coolant. The need for further experimental data in this area was one of the main reasons for having established the new experimental facilities at Winfrith. They provide the capability to carry out the experiments involving explosion under safe conditions by setting them up inside a strong test cell at the center of 200 m exclusive area. The instrumentation is provided for the transient measurements of pressure and strain and for high speed cinematography. The program concerned with the propagation of thermal interaction through a metal/water system was started in early 1975, and it is not yet completed. It is plausible to consider the development of large thermal interaction as a three stage process, namely mixing phase, the trigger process initiating thermal interaction, and the propagation of the interaction. With the rig THERMIR, the experiments on producing the violent interaction of tin and water and that of aluminum and water were carried out, and the characteristics of both interactions are reported. As the second stage of the Winfrith program, the UO_2/Na system will be studied, and this program is discussed.

- 2720 (PNC-N--251-76-12(Vol.1), pp 307-340)
EXPERIMENTAL STUDY ON CONVERSION OF STORED THERMAL ENERGY TO MECHANICAL WORK IN FUEL-COOLANT INTERACTION. Fujie, Y.; Kondo, S.; Miyazaki, K. (Osaka Univ., Suita (Japan). Faculty of Engineering). 1976.

Proceedings of the third specialist meeting on sodium/fuel interaction in fast reactors.

In the present study the process how the thermal energy stored in the coolant should be converted into the mechanical work is emphasized as well as the generation of pressure waves, eliminating the complicated problems involved in FCI, such as fragmentation, heat transfer from molten fuel to coolant, etc. A simulation experiment has been performed in a small scale vertical channel using stagnant potassium. Large superheat inherent in liquid metals was utilized as the method to accumulate the thermal energy. The following experimental parameters were chosen to examine their effects on the energy conversion ratio: the heat flux, the cover gas pressure, the liquid column length, the temperature of the upper unheated region and the incipient boiling superheat

(corresponding to the stored thermal energy). The results are summarized that the conversion ratio from thermal to kinetic energies has increasing trends with increases of the incipient boiling superheat, the cover gas pressure and the temperature of upper unheated region, ranging over 0.02-0.75% in the present experiment.

- 2721 (PNC-N--251-76-12(Vol.1), pp 179-186)
CURRENT STATUS OF EXPERIMENTAL AND THEORETICAL WORK ON SODIUM/FUEL INTERACTION (SFI) AT KARLSRUHE 'SIMULATION EXPERIMENTS'. Beutel, H.; Bojarsky, E.; Calderola, L.; Jacobs, H.; Reiser, H. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

Experimental and theoretical work on sodium fuel interaction (SFI) is being carried out at the Nuclear Research Center of Karlsruhe (Germany) within the framework of the 'Projekt Schneller Brueker (PSB)'. This paper concerns the description and the discussion of the experimental and theoretical results already obtained.

- 2722 (PNC-N--251-76-12(Vol.1), pp 363-380)
INJECTION OF SODIUM INTO LIQUID STAINLESS STEEL. A REPORT OF THE SECOND EXPERIMENT NA-SS/1. Abbey, M.J.; Asher, R.C.; Bradshaw, L.; Davies, D.; Sangwine, S.J. (UKAEA Research Group, Harwell. Atomic Energy Research Establishment). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

Also published in AERE-M--2771.

A second experiment in which liquid sodium was injected beneath the surface of molten steel has been carried out using an experimental technique based on that described in AERE - M 2770 but incorporating several improvements particularly in the instrumentation. The sodium (~ 1.5 g) was at $400 - 450^\circ\text{C}$ and the steel (stainless steel in this instance) was at $1750 - 1800^\circ\text{C}$. A vapour explosion occurred after a measured dwell time and two force pulses > 100 N and 60 N, separated by an interval of ~ 1 ms were transmitted to the base of the crucible containing the molten steel. Most of the sodium 'vapourized' but only a minor part of the steel was converted into small particles.

- 2723 (PNC-N--251-76-12(Vol.2), pp 839-875)
PHYSICAL MODEL AND CALCULATION CODE FOR FUEL COOLANT INTERACTIONS. Goldammer, H.; Kottowski, H. (Commission of the European Communities, Ispra (Italy). Joint Research Centre). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

A physical model is proposed to describe fuel coolant interactions in shock-tube geometry. According to the experimental results, an interaction model which divides each cycle into three phases is proposed. The first phase is the fuel-coolant-contact, the second one is the ejection of the coolant, and the third phase is the impact and fragmentation. Physical background of these phases are illustrated in the first part of this paper. Mathematical expressions of the model are exposed in the second part. A principal feature of the computational method is the consistent application of the fourier-equation throughout the whole interaction process. The results of some calculations, performed for different conditions are compiled in attached figures.

- 2724 (PNC-N--251-76-12(Vol.2), pp 561-593) MILLISECOND-PERIOD MELTDOWN EXPERIMENTS ON PROMPT-BURST EFFECTS AND MOLTEN-TIN-WATER DROPPING EXPERIMENTS. Wright, R.W. (Nuclear Regulatory Commission, Washington, D.C. (USA)); Coats, R.L.; Schmidt, T.R.; Arakeri, V.H. 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

The U.S. Nuclear Regulatory Commission has initiated a program of confirmatory research for the safety assessment of LMFBR plants. In the sodium-fuel interactions area, this research includes a series of real-time in-pile experiments on the pressure and work potential of prompt-burst excursions as well as laboratory dropping experiments with molten tin and water. The in-pile experiments are performed by Sandia Laboratories in the Annular Core Pulse Reactor (ACPR), which has a minimum period of 1.3 milliseconds. These single-pin experiments are performed in a piston-loaded, stagnant-sodium autoclave, that is conceptually similar to the one used in the S-11 TREAT test. Unlike the S-11 test, however, realistic radial temperature profiles are obtained in the fuel, the cladding, and the sodium by pre-pulsing the reactor about 1/2 second before the main pulse. A series of preparatory runs have been made with helium-filled capsules and at low energy with sodium-filled capsules. The first significant fuel-coolant interaction run is scheduled for late March 1976. This will be a double-pulsed run at 2700 J/gm UO_2 . A continuing series of experiments is planned with oxide and advanced fuels in both fresh and irradiated form. In molten-tin-water dropping experiments at UCLA, microsecond duration multi-flash photography has been used for event diagnostics. Transition or nucleate boiling was found to trigger energetic interactions or vapor explosions. Temperature stratification in the water was found to reduce the threshold tin temperature necessary to produce vapor explosions below that the predicted by the coolant homogeneous nucleation hypothesis. Interaction zone growth times of a few msec. were measured.

- 2725 (PNC-N--251-76-12(Vol.2)) PROCEEDINGS OF THE THIRD SPECIALIST MEETING ON SODIUM/FUEL INTERACTION IN FAST REACTORS. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). 1976. 479p. (CONF-760328--P2). Dep. NTIS (US Sales Only), PC A21/MF A01.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

Each paper has been separately processed for the Energy Data Base.

- 2726 (PNC-N--251-76-12(Vol.2), pp 651-680) ANALYSIS OF REACTOR STRUCTURAL RESPONSE TO FUEL SODIUM INTERACTION IN A HYPOTHETICAL CORE DISRUPTIVE ACCIDENT. Suzuki, K. (Nippon Atomic Industry Group Co. Ltd., Tokyo); Tashiro, M.; Sasanuma, K.; Nagashima, K. 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

This study shows the effect of constraints around FSI zone on FSI phenomena and deformations of reactor structures. SUGAR-PISCES code system has been developed to evaluate the phenomena of FSI and the response of reactor structure. SUGAR calculates the phenomena of FSI. PISCES, developed by Physics International Company in U.S.A., calculates the dynamic response of reactor structure in two-dimensional, time-dependent finite-difference Lagrangian model. The results show that the peak pressure and energy by FSI and the

deformation of reactor structures are about twice in case of FSI zone surrounding by blanket than by coolant. The FSI phenomena highly depend on the reactor structure and the realistic configuration around core must be considered for analyzing hypothetical core disruptive accident. This work was supported by a grant from Power Reactor and Nuclear Fuel Development Corporation.

- 2727 (PNC-N--251-76-12(Vol.2), pp 751-783) INVESTIGATIONS ON FUEL-COOLANT INTERACTION WITH THE RCN CODE VS-4. Van der Putten, A.P.W.M.; Koning, H.; Van den Bogaard, J.P.A. (Reactor Centrum Nederland, Petten). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

A set of partial differential equations describing the conservation of mass, energy and momentum in a coolant channel as a function of axial coordinate and time are solved with the RCN Code VS-4, using the method of characteristics. The viscosity and turbulence of coolant are taken into account in the usual manner of technical thermodynamics by the introduction of a friction coefficient in the momentum equation. The heat transported by the axial mass flow in the subassembly is assumed to be so large that the heat transport by axial conduction may be neglected. The heat transport by conduction in the radial direction is assumed to be perfect. The calculation results are discussed and concluded as follows. (1) The method of characteristics was proved to be applicable not only to the one phase but also to the description of the homogeneous equilibrium two phase part of a thermohydraulic process. (2) To investigate the reactor safety aspects of FCI (fuel-coolant interaction) process and to evaluate the experimental research on FCI, it is indispensable to take the axial dependence effects into account. To this end, the axially dependent code VS-4 based on the method of characteristics is available. (3) At present, the thermodynamic properties of sodium according to Golden and Tokar are used in the program. To improve the accuracy of the code for high temperature and pressure, a consistent set of thermodynamic properties of sodium must be incorporated in the program.

- 2728 (PNC-N--251-76-12(Vol.2), pp 819-837) CURRENT STATUS OF EXPERIMENTAL AND THEORETICAL WORK ON SODIUM/FUEL INTERACTION (SFI) AT KARLSRUHE 'CODE DEVELOPMENTS'. Beutel, H. (Kernforschungszentrum Karlsruhe (Germany, F.R.G.). Inst. fuer Reaktorentwicklung); Bojarsky, E.; Caldarola, L.; Jacobs, H.; Reiser, H. 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

Two computer codes have been developed. The first code contains a heat transfer model based on the inverse Leidenfrost phenomenon. The exact solution of the thermal diffusion equation for a sphere is included in the code. The time history of each fuel particle, the presence of fission gas, the size distribution of fuel particles, the effect of the friction due to channel walls, and the effect of the pressure loss at channel outlet can be taken into account. Extensive parametric study has been carried out with this code, and the main conclusions are summarized. The second code is used for the analysis of propagation effect. The interaction region can be subdivided into arbitrary number of sections, each containing fuel and coolant. The thermal conductivity of liquid sodium and the presence of non-condensing gas can be taken into account. The numerical results are generally consistent with those obtained with the first code. Vapor

bubbles and their collapse seem to be the most probable mechanism of the fragmentation of UO_2 in sodium. For this reason, work has been focused on modeling this mechanism, and a preliminary model was developed. The first theoretical model for FCI developed at Karlsruhe was coupled to the American REXCO code which is available at Ispra, and the FCI subroutine is already available.

2729 (PNC-N--251-76-12(Vol.2), pp 785-817)

MODEL FOR FUEL-SODIUM INTERACTION.

APPLICATION TO THE JEF EXPERIMENTS. Breton, J.-P.; Antonakas, D. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

A model of sodium-fuel interaction, referred to as TRACONA-BUEE, has been developed. The fuel particles are assumed to be introduced in the interacting zone within a finite mixing time, according to a given function (not necessarily linear). The equations for heat transfer inside fuel particles are those of Cho and Wright (transient conduction for phase A and quasi-steady state heat transfer for phase B). During phase B several options for heat transfer from fuel to sodium can be assumed (no transfer, transfer proportional to the volume fraction of liquid sodium, given duration of transfer, etc...). Two versions are available: a spherical one (EPISCOPUS) and an axial one (TEXAS). For application to the JEF experiments a model of heat losses along the cold column had to be introduced into TEXAS. It was found that the phenomenon is essentially governed by the heat losses. The velocity of the cold sodium in the column presents marked maxima and minima. The agreement with experiment is satisfactory.

2730 (PNC-N--251-76-12(Vol.2), pp 507-543)

INTERACTIONS WITH SMALL AND LARGE SODIUM TO UO_2 MASS RATIOS. Clerici, G.; Holtbecker, H.; Schins, H.; Schlittenhardt, P. (Commission of the European Communities, Ispra (Italy). Joint Research Centre). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

This paper presents final results of the Ispra dropping experiments. After a brief explanation of the alteration of the experimental facility the results of experiments and their theoretical interpretations are presented. And it is concluded that the test have shown that coherent energetic explosions cannot be produced by dropping small amounts of molten UO_2 into large quantities of sodium. The second, the study of sodium entrapment is described. An experimental device to inject sodium in a UO_2 filled crucible is explained. The third part briefly describes two computer codes, Press 1 and 2. In the first one, the constraint supplied by the solidifying UO_2 shell around a sodium drop is evaluated. In the second one, a liquid-liquid contact between UO_2 and sodium is considered. Some numerical evaluations were made with these codes. Results of the evaluations of the mechanical constraint due to the solidifying UO_2 shell, and the heating and expansion of sodium in UO_2 are presented.

2731 (PNC-N--251-76-12(Vol.2), pp 877-890)

CALCULATIONS OF THE POSSIBLE CONSEQUENCES OF MOLTEN FUEL SODIUM INTERACTIONS IN SUBASSEMBLY AND WHOLE CORE GEOMETRIES. Coddington, P.; Fishlock, T.P.; Jakeman, D. (UKAEA Reactor

Group, Winfrith. Atomic Energy Establishment). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

The possible consequences of molten fuel sodium interactions are calculated using various modelling assumptions and key parameters. And the significance of the choice of assumptions and parameters are discussed. As for subassembly geometry, the results of one-dimensional code EXPEL are compared with the solutions of the one-dimensional Lagrangian equations of a compressible fluid (TOPAL was used). The adequacy of acoustic approximation used in EXPEL is discussed here. The effects of heat transfer time constant on the behaviour of peak pressure are also analyzed by parametric surveys. Other items investigated are the length and position of the interacting zone, the existence of a non-condensable gas volume, and the vapour condensation on cold clad. As for whole core geometry, a simple dynamical model of arc expanding spherical interacting zone immersed in a semi-infinite sea of cold liquid was used (SHORE code). Within the interacting zone a simple heat transfer model (including a heat transfer time and a fragmentation time) was adopted. Vapour blanketing was considered in a number of ways. Representative results of the calculations are given in a table. Containment studies were also performed for 'ducted' design and 'open pool' design. The development of new codes in the U.K. for these analysis are also briefly described.

2732 (PNC-N--251-76-12(Vol.2), pp 623-650)

REVIEW OF FRAGMENTATION MODELS RELATIVE TO MOLTEN UO_2 BREAKUP WHEN QUENCHED IN SODIUM COOLANT. Cronenberg, A.W. (New Mexico Univ., Albuquerque (USA)); Grolmes, M.A. 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

An important aspect of the fuel-coolant interaction problem relative to liquid metal fast breeder reactor (LMFBR) safety analysis is the fragmentation of molten oxide fuel during contact with liquid sodium coolant. A proper description of the kinetics of such an event requires an understanding of the breakup process and an estimate of the size and dispersion of such finely divided fuel in coolant. In recent years, considerable interest has centered on the problem of determining the nature of such fragmentation. In this paper, both analytic and experimental studies pertaining to such breakup are reviewed in light of recent developments in the understanding of heat transfer and solidification phenomena during quenching of UO_2 in sodium. A more extensive review of this subject can be found in Ref. 1.

2733 (PNC-N--251-76-12(Vol.2), pp 733-750)

REVIEW OF THE MEETINGS OF THE WORKING GROUP ON THE COMPARISON OF CALCULATION MODELS.

Jakeman, D. (UKAEA Reactor Group, Winfrith. Atomic Energy Establishment). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

Physical models used in various SFI analysis codes are reviewed and the results of the comparison of productions when the codes were applied to specific test cases are presented. Several representative codes are taken up for comparisons. The features that are compared and discussed in this review paper are; the fragmentation of molten fuel; heat transfer for Stage A (single phase coolant); heat transfer for Stage B (two phase coolant); hydrodynamics; and equation of the state of sodium. A brief description of the interaction conditions

applied for the test cases is given followed by the results of comparisons. Instantaneous fuel fragmentation time of 10 ms. were assumed throughout the comparison calculations. The items being compared and discussed are the peak pressure and work done in Stage A, peak pressure in Stage B, time for expulsion of sodium from channel, and the work done in expelling sodium from channel. Results of comparisons of calculated values are presented in tables and figures. One of the main conclusions drawn from these comparisons is that if expulsion times are the main interest the various treatments adopted in different codes do not lead to very different predictions.

- 2734 (PNC-N--251-76-12(Vol.2), pp 681-732) PARAMETRIC STUDY ON REACTOR CONTAINMENT RESPONSE TO FUEL-SODIUM INTERACTION. Sawada, T.; Hasegawa, Y.; Nishimura, A. (Mitsubishi Atomic Power Industries, Inc., Tokyo (Japan)). 1976.

From 3. specialists' meeting on sodium fuel interaction; Tokyo, Japan (Mar 1976).

Evaluations have been made on the effects of FSI parameters which affect dynamic responses of reactor containment structures. A coupled computer code FCI-II/ASPRIN-II has been employed in this study. The FCI-II code is based on a modified rate-limited model originally developed by Cho, Ivins and Wright. The ASPRIN-II code is a modified version of the reactor containment response analysis code ASPRIN originally developed by G. L. Fox. Coupling between an FSI region and its surroundings is determined by FSI zone pressure, $P(V,t)$, as well as by FSI zone volume which is a function of constraint conditions. A spherical FSI zone in the core of a demonstration plant is assumed in this study. 1000Kg of molten fuel at 3338°K is mixed with 105Kg of sodium coolant at 900°K. The parameters changed are constraint conditions and FSI parameters such as a fuel particle size, a rate of fragmentation and mixing, a fuel-sodium mixing ratio and a two-phase heat transfer coefficient at a fuel particle surface.

- 2735 (PNC-N--251-76-26) PROGRESS REPORT ON FAST BREEDER REACTOR DEVELOPMENT IN JAPAN. OCTOBER 1975--JANUARY 1976. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). May 1976. 8p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Following the completion of the in-argon, high temperature test, the in-sodium functional test of Joyo has set in. The fabrication of the equipments for monitoring the flow rate and temperature in the center channel and the power distribution was finished. The modification of design of the prototype fast breeder reactor Monju came into its phase 3. The interim report on the check-up and review of the Monju project by the Government is now ready. Various calculation codes were developed or are in development stage. The mock-up assembly FCA 7-1 has been built, which consists of the Pu-fueled sector region simulating the Monju core and the U-235-fueled driver region. Various reactor physics experiments have been carried out in this assembly. Also, the calculation methods for reactor physics parameters have been developed, and the detailed calculation on the main shield of Joyo was performed. The situation of the developments of the components for Joyo and Monju and the measuring and control systems is shown. Almost all the existing sodium test facilities in Oarai Engineering Center were in service without any trouble, and the new test facility named

'Carbon transfer test loop' was commissioned. The progress in the fields of sodium technology, fuel and material, safety and steam generators is reported.

- 2736 (PNC-N--251-78-02) PROGRESS REPORT ON FAST BREEDER REACTOR DEVELOPMENT IN JAPAN, OCTOBER - DECEMBER 1977. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Jul 1978. 12p. Dep. NTIS (US Sales Only), PC A02/MF A01.

As for the development of the experimental fast breeder reactor 'Joyo', the low power performance test was carried out, and the sodium void effect, power distribution and others were measured. It was finished on November 19, 1977. Thereafter, the work of installing a pile oscillator started, while the periodical inspection was under way. The maintenance work for the transfer rotor has been completed. Fabrication of the pump maintenance cask, construction of the spent fuel storing and cooling facility and others are continued. The operation, maintenance and inspection of various systems were made in preparation for the reactor performance test. The examination on the safety of the reactor for power increase up to 100 MW is continued. The core characteristic analysis is in progress. The construction preliminary design (1) of the prototype fast breeder reactor 'Monju' is being carried out. For the analysis and preparation of data to be submitted for the safety licensing, the analysis of decay heat (2) was finished, and the draft of application for the installation of the reactor plant has been completed. The development of various calculation codes is in progress. The preliminary design studies on the demonstration reactor (3) are continued. The situations of various researches and developments on fast breeder reactors are described.

- 2737 (PNC-N--952-76-04) OPERATION MANUAL FOR SODIUM-MEASURING EQUIPMENT TEST LOOP. Ooyama, N.; Nakamoto, K.; Sekiguchi, N.; Nakasuti, T.; Atsumo, H. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Jan 1976. 190p. (In Japanese). Dep. NTIS (US Sales Only), PC A10/MF A01.

This operation manual contains the materials and references required to operate and maintain the sodium-measuring equipment test loop that was built for the test in sodium of in-core monitoring instrumentation for the fast breeder prototype reactor 'Monju'. The test loop consists of sodium equipment and piping installation, gas and vacuum equipment and piping installation, preheat and heat insulation facilities, instrumentation and control equipments, power supply unit, rack and stand installation and others. It uses liquid metal sodium of approximately 2900 kg as the fluid and high purity argon gas as the cover gas. Maximum sodium temperature is specified at 600 deg C, maximum sodium flow is approximately 1000 l/min each in main circulation and auxiliary systems, and maximum working pressure is 5 kg/cm² G (normally 2 kg/cm² G). The procedures of start-up, operation and shut down are described, and the measures in abnormal and emergency conditions, maintenance and checking-up, electrical instrumentation equipments, and sodium, gas and vacuum systems are explained in detail. The last two sections are included for the reference because they are necessary for operation and maintenance, but are not used very often.

- 2738 (PNCT--831-75-02, pp 42-52)

DETERMINATION OF O/M RATIOS OF UO_2 - PuO_2 MIXED OXIDE FUEL, (1). STATISTICAL ANALYSIS OF O/M RATIOS. Sone, T.; Okamoto, H.; Isagawa, H.; Ochiai, Y.; Kano, K. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Dec 1975.

Semi-annual progress report of Power Reactor and Nuclear Fuel Development Corporation, Tokai Works. Jan--Jun 1975.

Statistical analysis of variance was made for the determination of the O/M ratio of MONJU fuel pellets. Two procedures were used for this purpose. One is the experimental layout procedure with ramification, and the other is the two-way layout procedure with repetition. The experimental methods used for the determination of O/M ratio are presented in the first part. Two mechanically blended pellets taken from one lot in a production line were crushed and divided into five parts. Three of them were analyzed on the first day, and the remaining two parts were kept in a desiccator for analysis on another day. The procedures of the analysis of variance are explained in the second part. The production lot, the date of analysis, and the divided parts were selected as the variation factors of the measured O/M ratio. Two procedures of analysis are explained and summarized in tables. The measured results of O/M ratio and the results of analysis by two procedures are presented in tables. Some discussion is given in the third part, based on those results.

2739 (PNCT--831-77-01, pp 95-99) DEVELOPMENT OF NONDESTRUCTIVE MEASUREMENT OF COLD WORK RATE, (1). Tsunoda, N.; Ando, H.; Kamimura, H.; Inada, E. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Apr 1977.

Semi-annual progress report of Power Reactor and Nuclear Fuel Development Corporation, Tokai Works, January--June 1976.

Cold-worked stainless steel type 316 is used as the cladding tube material for the prototype FBR Monju in Japan. The in-reactor life term of cold-worked stainless steel is longer than that of solution-treated, but there is no method of measuring cold work rate at present. In order to develop the nondestructive method for measuring the distribution of cold work rate over the length of cladding tubes, a study was made on the method of measuring the residual magnetism in cladding tubes. The test specimen was a cladding tube made of the same material and by the same process as those for Monju. The block diagram for the system measuring the residual magnetism is shown. First, the tube is demagnetized by rotating it in strong alternating magnetic field. Next, the tube is linearly moved without rotation through strong unidirectional magnetic field and magnetized. Then the tube is rotated at constant speed above the magnetometer probe, and the retained field is detected. The measurement was made at nine locations. The residual magnetism of the specimen was constant when a magnetizing force over 2000 gauss was applied. The high degree of correlation was shown between the residual magnetism and the cold work rate. Therefore, it is possible to know cold work rate by measuring residual magnetism.

2740 (PNCT--831-77-02, pp 27-41) INSPECTING EXAMINATION OF MOCKUP FUEL ASSEMBLIES OF MONJU AFTER SODIUM FLOW TEST. Miura, M.; Ando, H.; Taki, K.; Kitano, M. (Power Reactor and Nuclear Fuel Development Corp., Tokai, Ibaraki (Japan)). Tokai Works. Sep 1977.

Semi-annual progress report of Power Reactor and Nuclear Fuel Development Corporation, Tokai Works. Jul. - Dec. 1976.

As a part of the out-of-pile examinations of fuel assemblies for the prototype fast breeder reactor MONJU, the sodium flow test of mockup assemblies was performed. Each tested fuel assembly contains 166 - 169 fuel pins, which are 6.5 mm in diameter, arranged in triangular array, and wrapped in a hexagonal wrapper tube. The total number of tested fuel assemblies are 9; 4 of wire wrapped type, 4 of honeycomb grid type and 1 of rhombus grid type. The sodium flow test was conducted under the conditions of 600°C (temperature), 524 - 4800 hr (time), and 2 - 60 ppm (oxygen content in sodium). After this flow test, the fuel assemblies were dismantled, and the fuel pin bundles were inspected by visual inspection and pin gap measurement, and the fuel pins were inspected by helium leak test, weight measurement, micrograph of external condition, surface roughness measurement, total length measurement, test of looseness of wire, bending measurement and measurement of axial wire wrapping pitch. Some increase of surface roughness, minute form change, and others have been recognized as the result of these various nondestructive tests.

2741 (PNL--2413) ANALYSIS OF THE IMPACT OF RETRIEVABLE SPENT FUEL STORAGE. Merrill, E.T.; White, M.K.; Fleischman, R.M. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1978. Contract EY-76-C-06-1830. 19p. Dep. NTIS, PC A02/MF A01.

The impact of retrievably storing spent fuel is measurable in terms of the contribution the stored spent fuel makes to implementing the fuel management option selected. For the case of a decision to recycle LWR fuel in LWRs, a useful indicator of impact is the ratio of energy production with varying degrees of spent fuel retrievability to that achievable with total spent fuel retrievability. For a decision made in the year 2000, this ratio varies from 0.81 (10 yr storage in reactor basins) to 0.97 (retrievable storage for 25 years after fuel discharge). An earlier decision to recycle in LWRs results in both of these ratios being nearer to 1.0. If a decision is reached to implement a breeder reactor economy, the chosen comparison is the installed breeder capacity achievable with varying degrees of spent fuel retrievability. If a decision to build breeder reactors is reached in the year 2000, the maximum possible installed breeder capacity in 2040 varies from 490 GWe (10 yr storage in reactor basins) to 660 GWe (all fuel retrievably stored). If all fuel is retrievably stored 25 years, 635 GWe of breeder capacity is achievable by 2040. For an earlier decision date, such as 1985, the maximum possible installed breeder capacity in 2040 ranges from 740 GWe (no retrievable storage) to 800 GWe (all fuel retrievably stored). As long as a decision to reprocess is reached before 2000, most of the potential benefit of retrievable storage may be realized by implementing retrievable storage after such a decision is made. Neither providing retrievable spent fuel storage prior to a decision to reprocess, nor designing such storage for more than 25 years of retrievability appear to offer significant incremental benefit.

2742 (PNL--2417) FRACTURE MECHANICS OF THIN WALL CYLINDRICAL PRESSURE VESSELS: AN INTERIM REVIEW. Kurtz, R.J.; Olson, N.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Aug 1977. Contract EY-76-C-06-1830. 40p. Dep. NTIS, PC A03/MF A01.

The report is a result of activities in the LMFBR Fuel Rod Transient Performance Program sponsored by the LMFBR Branch of the Division

of Project Management, U.S. Nuclear Regulatory Commission. One of the objectives is to develop predictions relative to the length, direction, and rate of growth of cladding rips subsequent to (or concurrent with) the initial cladding breach during unprotected transients. To provide a basis for evaluation, Battelle, Pacific Northwest Laboratories has reviewed most available fracture mechanics assessments relative to thin-wall cylindrical pressure vessels. The purpose of the report is to review the various fracture mechanics models and to describe the pertinent fracture parameters. It is intended to provide a formal basis for assessing future analytical predictions of fracture behavior of materials exposed to transient LMFBR thermal and mechanical loading conditions. In addition, the report is expected to provide reference material for evaluating or developing experimental programs required to properly address the problem of predicting fracture behavior of materials during transient events.

- 2743 (PNL--2447) EVALUATION OF LIQUID METAL LEAK DETECTION METHODS FOR THE CLINCH RIVER BREEDER REACTOR PLANT. Morris, C.J.; Doctor, S.R. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1977. Contract EY-76-C-06-1830. 111p. Dep. NTIS, MF A01.

This report documents an independent review and evaluation of sodium leak detection methods described in the Clinch River Breeder Reactor Preliminary Safety Analysis Report. Only information in publicly available documents was used in making the assessments.

- 2744 (PNL--2500(Pt.1), pp 3.55-3.58) CHARACTERIZATION OF MIXED LMFBR FUEL-SODIUM AEROSOLS GENERATED BY LASER VAPORIZATION. Allen, M.D. Feb 1978.

Pacific Northwest Laboratory annual report for 1977 to the DOE Assistant Secretary for Environment. Part 1. Biomedical sciences.

A laser was used to vaporize $\text{PuO}_2\text{-UO}_2$ from the surface of a rotating fuel pellet to simulate LMFBR fuel-sodium condensation aerosols. The branched, chain-like $\text{PuO}_2\text{-UO}_2$ aggregates are swept by argon into a sodium vapor atmosphere. The sodium condenses on the aggregates, and the mixed aerosol is diluted with 50% relative humidity air. These aerosols are being characterized according to physical appearance, crystalline structure, and aerodynamic behavior.

- 2745 (PNL--2500(Pt.1), pp 3.59-3.60) TRANSLOCATION OF MIXED LMFBR FUEL-SODIUM AEROSOLS FROM THE LUNG FOLLOWING INHALATION BY RODENTS. Mahlum, D.D.; Hess, J.O.; Allen, M.D. Feb 1978.

Pacific Northwest Laboratory annual report for 1977 to the DOE Assistant Secretary for Environment. Part 1. Biomedical sciences.

Rats were exposed to aerosols consisting of either LMFBR fuel or mixtures of sodium and fuel. Pulmonary deposition of the mixed aerosol was usually less than that of the fuel only because of the larger size of the mixed aerosol. When the ratio of sodium to fuel was high (>32), the rate of clearance from the lung was faster than for the aerosols with low sodium-to-fuel ratios.

- 2746 (PNL--2805) ANALYSIS OF VARIOUS NDT TECHNIQUES TO DETERMINE THEIR FEASIBILITY FOR DETECTING THIN LAYERS OF FERRITE ON TYPE 316 STAINLESS STEEL. Dudder, G.B.; Atteridge, D.G.; Davis, T.J. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Sep 1978.

Contract EY-76-C-06-1830. 30p. Dep. NTIS, PC A03/MF A01.

The applicability of various NDT techniques for detecting thin layers of ferrite on Type 316 stainless steel cladding was studied. The ability to detect sodium-induced ferrite layers on fuel pins would allow an experimental determination of the fuel pin temperature distribution. The research effort was broken down into three basic sections. Phase one consisted of a theoretical determination of the ferrite detection potential of each of the proposed NDT techniques. The second phase consisted of proof-of-principle experiments on the techniques that passed phase one. The third phase consisted of in-hot cell testing on actual EBR-II fuel pins. Most of the candidate techniques were eliminated in the first phase of analysis. Four potential techniques passed the initial phase of analysis but only three of these passed the second analysis phase. The three techniques that passed the proof-of-principle section of analysis were heat tinting, magnetic force and electromagnetic techniques. The electromagnetic technique was successfully demonstrated on actual fuel pins irradiated in EBR-II in the third phase of analysis while the other two techniques were not carried to the hot cell analysis phase. Results of this technique screening study indicates that an electromagnetic and/or heat tinting ferrite layer NDT technique should be readily adoptable to hot cell inspection requirements. It was also concluded that the magnetic force technique, while feasible, would not readily lend itself to hot cell fuel pin inspection.

- 2747 (PNL--2850(Pt.1), pp 3.34-3.46) DISSOLUTION OF LMFBR FUEL-SODIUM AEROSOLS. Allen, M.D.; Moss, O.R. Feb 1979.

Pacific Northwest laboratory annual report for 1978 to the DOE Assistant Secretary for Environment. Part 1. Biomedical sciences.

Plutonium dioxide, normally insoluble in biological fluids, becomes much more soluble when mixed with sodium as the aerosol is formed. Sodium-fuel aerosols are approximately 20 times less soluble in simulated lung fluid than in distilled water. Solubility of sodium-fuel aerosols increases when Na_2CO_3 are added to the distilled-water dissolution fluid. Mixed-oxide fuel aerosols without sodium present are relatively insoluble in distilled water, simulated lung fluid, and distilled water with Na_2CO_3 and NaHCO_3 added.

- 2748 (PNL--2850(Pt.1), pp 3.47) SPECTROPHOTOMETRIC ANALYSES OF LMFBR FUEL-SODIUM AEROSOLS. Morgan, L.G.; Ryan, J.L.; Allen, M.D. Feb 1979.

Pacific Northwest laboratory annual report for 1978 to the DOE Assistant Secretary for Environment. Part 1. Biomedical sciences.

Spectrophotometric investigations of $(\text{Pu,U})\text{O}_2$, sodium- $(\text{Pu,U})\text{O}_2$, and sodium- PuO_2 aerosols show that enhanced solubility of uranium and plutonium occurs as a result of interaction of sodium with either mixed-oxide or PuO_2 aerosols, with or without exposure to air. Results indicate that uranium is oxidized in the presence of sodium and air and is soluble as sodium uranyl tricarboxylate. Plutonium is soluble as a Pu(IV) species from aerosols exposed to sodium and air. No evidence of a soluble Pu(VI) species has been found.

- 2749 (PNL--2850(Pt.1), pp 3.48-3.50) BIOLOGICAL BEHAVIOR OF MIXED LMFBR-FUEL-SODIUM AEROSOLS. Mahlum, D.D.; Hackett, P.L.; Hess, J.O.; Allen, M.D. Feb 1979.

Pacific Northwest laboratory annual report for 1978 to the DOE Assistant Secretary for Environment. Part 1. Biomedical sciences.

Immediately after exposure of rats to mixed aerosols of sodium-LMFBR fuel, about 80 to 90% of the body burden of ^{239}Pu is in the gastrointestinal tract; 1.5 to 4% is in the lungs. With fuel-only aerosols, less of the body burden was in the GI tract and more in the lung and the head. Blood and urine values suggest an increased absorption of ^{239}Pu from sodium-fuel than from fuel-only aerosols.

- 2750 (PNL--2923) COBRA-IV COMPARISON WITH THE WARD 11:1 SCALE LMFBR AIR FLOW TESTS. Basehore, K.L.; George, T.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Mar 1979. Contract EY-76-C-06-1830. 61p. Dep. NTIS, PC A04/MF A01.

The COBRA-IV code was verified against the results of the Westinghouse Advanced Reactor Division (WARD) 11:1 scale liquid metal fast breeder reactor (LMFBR) fuel assembly air flow tests. The major effort in comparing COBRA results with the data was directed toward matching the transverse velocities, though axial velocity comparisons were also made. Four input parameters to the COBRA code are known to have a significant effect on the velocity distribution within a wirewrapped LMFBR bundle. The relevant parameters are: the turbulent mixing coefficient, the turbulent crossflow resistance factor, the effective pitch length for forcing crossflow and the transverse momentum geometry factor. Computational results for various values of each of these input quantities are presented for the transverse velocity component of the flow field. Axial velocity comparisons are also made using the calibrated set of parameters.

- 2751 (PNL--2938) COBRA-IV WIRE WRAP DATA COMPARISONS. Donovan, T.E.; George, T.L.; Wheeler, C.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Feb 1979. Contract EY-76-C-06-1830. 42p. Dep. NTIS, PC A03/MF A01.

Thermal hydraulic analyses of hexagonally packed wire-wrapped fuel assemblies are complicated by the induced crossflow between adjacent subchannels. The COBRA-IV computer code simultaneously solves the hydrodynamics and thermodynamics of fuel assemblies. The modifications and the results are presented which are predicted by the COBRA-IV calculation. Comparisons are made with data measured in five experimental models of a wire-wrapped fuel assembly.

- 2752 (PNL--3129) ENVIRONMENTAL CONTROL ASPECTS FOR FABRICATION, REPROCESSING AND WASTE DISPOSAL OF ALTERNATIVE LWR AND LMFBR FUELS. Nolan, A.M.; Lewallen, M.A.; McNair, G.W. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Nov 1979. Contract EY-76-C-06-1830. 94p. Dep. NTIS, PC A05/MF A01.

Environmental control aspects of alternative fuel cycles have been analyzed by evaluating fabrication, reprocessing, and waste disposal operations. Various indices have been used to assess potential environmental control requirements. For the fabrication and reprocessing operations, 50-year dose commitments were used. Waste disposal was evaluated by comparing projected nuclide concentrations in ground water at various time periods with maximum permissible concentrations (MPCs). Three different fabrication plants were analyzed: a fuel fabrication plant (FFP) to produce low-activity uranium and uranium-thorium fuel rods; a plutonium fuel

refabrication plant (PFRFP) to produce plutonium-uranium and plutonium-thorium fuel rods; and a uranium fuel refabrication plant (UFRFP) to produce fuel rods containing the high-activity isotopes ^{232}U and ^{233}U . Each plant's dose commitments are discussed separately. Source terms for the analysis of effluents from the fuel reprocessing plant (FRP) were calculated using the fuel burnup codes LEOPARD, CINDER and ORIGEN. Effluent quantities are estimated for each fuel type. Bedded salt was chosen for the waste repository analysis. The repository site is modeled on the Waste Isolation Pilot Program site in New Mexico. Wastes assumed to be stored in the repository include high-level vitrified waste from the FRP, packaged fuel residue from the FRP, and transuranic (TRU) contaminated wastes from the FFP, PFRFP, and UFRFP. The potential environmental significance was determined by estimating the ground-water concentrations of the various nuclides over a time span of a million years. The MPC for each nuclide was used along with the estimated ground-water concentration to generate a biohazard index for the comparison among fuel compositions.

- 2753 (PNL--3249) MASS FLOWS FOR LMFBRs FUELED WITH VARIOUS TYPES OF PLUTONIUM AND FERTILE MATERIAL. Jenquin, U.P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1979. Contract EY-76-C-06-1830. 39p. Dep. NTIS, PC A03/MF A01.

Results are presented of neutronics analyses performed to determine fueling requirements and mass flows for a typical 1200 megawatt electric (MWe) liquid metal fast breeder reactor (LMFBR). In performing the analyses, three plutonium compositions were considered to fuel the reactor. The composition depended on the amount of irradiation either a UO_2 or $\text{PuO}_2\text{-ThO}_2$ fuel had received in a light water reactor (LWR). Uranium and thorium fertile materials were considered. Calculations were performed to determine the cross sections, reactivity, and burnup for each fuel-fertile material configuration. Based on these calculations, the mass flows, plutonium enrichment requirements, and breeding ratios for each configuration were determined.

- 2754 (PNL--3255) DEPENDENCE OF LMFBR FUELING REQUIREMENTS ON PLUTONIUM ISOTOPIC COMPOSITION. Jenquin, U.P.; Newman, D.F. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Dec 1979. Contract EY-76-C-06-1830. 45p. Dep. NTIS, PC A03/MF A01.

When the liquid metal fast breeder reactors (LMFBRs) start up, plutonium from thermal reactors will be required to enrich the reactor fuel. Neutronics analyses performed to determine the fueling requirements for an 800 megawatt electric (MWe) breeder reactor are described. This study was initiated when it was assumed that thermal reactor plutonium recycle would take place prior to the introduction of the breeder. Therefore, a range of plutonium compositions was fed into the LMFBR and the fueling requirements were determined for these various compositions. Depleted uranium was used as the diluent. The analysis was performed for two reactor designs - a reference small fuel pin design and an advanced large fuel pin design. Burnup calculations were performed to determine the compositions of the discharge plutonium.

- 2755 (PNL-8A--7503) INITIAL STAGE RESTRUCTURING IN SPHERE-PAC MIXED-CARBIDE FUEL. Guenther, R.J.; Peddicord, K.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)).

1979. Contract EY-76-C-06-1830. 10p.
(CONF-790306--25). Dep. NTIS, PC A02/MF A01.
From International conference on fast breeder reactor performance; Monterey, CA, USA (5 Mar 1979):
The analysis of sintering models and mechanisms for mixed-carbide sphere-pac fuel has shown that volume diffusion is the dominant mechanism. The actual diffusion path is not clearly defined but the importance of small pressures in increasing neck growth is apparent. The time dependence of the neck ratios indicates that significant restructuring occurs within 5.6 hours which may be used as a bench mark for the beginning of pore migration.
- 2756 (PNL-SA--7521) CORTAN CODE FOR TEMPERATURE PREDICTIONS IN AN LMFBF DURING A NATURAL CIRCULATION TRANSIENT. Khan, E.U.; Hinds, A. (Battelle Pacific Northwest Labs., Richland, WA (USA); Argonne National Lab., IL (USA)). Jan 1979. Contract EY-76-C-06-1830. 6p. (CONF-790602--92). Dep. NTIS, PC A02/MF A01.
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
Three transient thermal-hydraulic models, TRAN I, II, III, have been incorporated into the CORTAN code. The TRAN I, II, III models are similar in as much as they retain the wire-wrap mixing and swirl flow models of the Energy code, i.e., the wire-wrap mixing is modeled by an enhanced eddy diffusivity in the central regions of the bundle and a swirl flow in the region near the duct wall. The fuel rod and duct wall models required to treat the stored heat and thermal inertia effects, are also similar in the three models. Up to 10 radial nodes are allowed in the rods and 6 radial nodes in the duct wall to solve the one-dimensional conduction equation for the rods and duct wall, by finite difference. Temperature-dependent properties are taken into account in calculating rod temperatures.
- 2757 (PNL-SA--7878) ANALYSIS OF STEADY STATE COMBINED FORCED AND FREE CONVECTION DATA IN ROD BUNDLES. Juneau, J.; Khan, E.U.; Cheatham, R.L. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Jun 1979. Contract EY-76-C-06-1830. 6p. (CONF-791103--75). Dep. NTIS, PC A02/MF A01.
From American Nuclear Society meeting; San Francisco, CA, USA (12 Nov 1979).
Fuel and blanket assemblies in an LMFBF are subjected to a wide range of power and power gradients during their life in the reactor. To accommodate these changes the assemblies operate in a wide range of flow regimes extending from forced convection, turbulent flow, to mixed convection, laminar flow. At low flow conditions the transverse temperature gradient in an assembly is considerably flattened because of energy redistribution by not only wire-wrap mixing and thermal conduction but also by flow redistribution because of buoyancy-induced crossflow. This has significance in LMFBF design. For the mixed convection regime of bundle operation, the transverse velocity profiles within a bundle change axially because of buoyancy-induced crossflow. It was therefore decided to use the ENERGY II and ENERGY III computer programs for the analysis of the rod bundle mixed convection data.
- 2758 (PP--04-PE-001) REMOVAL OF RESIDUAL SODIUM FROM AI MODULAR STEAM GENERATOR (SCTI-SP-312). (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 6 Nov 1973. Contract EY-76-C-03-0700. 35p. AT.
Portions of document are illegible.
Process procedures are enumerated for the removal of residual sodium from the AI modular steam generator prior to its removal from the Sodium Components Test Installation at the Liquid Metals Engineering Center. (DG)
- 2759 (R--2267-DOE) EXAMINATION OF ALTERNATIVE NUCLEAR BREEDING METHODS. Dreyfuss, D.J.; Augenstein, B.W.; Mooz, W.E.; Sher, P.A. (RAND Corp., Santa Monica, CA (USA)). Jul 1978. Contract EX-76-C-01-2337. 126p. Dep. NTIS, PC A07/MF A01.
This report presents a preliminary evaluation of the economics of using external source neutrons (provided by a DT fusion device or by a high-energy accelerator providing a proton or deuteron particle beam) for breeding fissile fuel, and compares these costs with those of the most intensively investigated reactor breeder (or internal neutron source breeder), the liquid metal fast breeder reactor (LMFBF). A simple evaluation model is used that calculates the net present discounted value, using a 10 percent discount rate of the cost to satisfy a specific demand for electricity over the period to 2050. Present discounted values of costs are estimated for four energy technologies: the uranium-fueled light water reactor (LWR), the LMFBF, the fusion hybrid breeder, and the accelerator-driven breeder. The latter two technologies produce fissile fuel which is then consumed in ordinary converter reactors. The discounted costs to produce electrical energy using the three breeding technologies to satisfy this demand are calculated and compared to a standard or base case where the LWR satisfies the demand. The cost differences between the base case and the three alternatives are compared to estimate the savings possible over the LWR base case. The conclusion is that the fusion hybrid breeder and the accelerator breeder, given our present information about the various technologies, promise to be competitive with reactor-based breeders such as the LMFBF and offer a number of qualitative advantages as well.
- 2760 (RAND/N--1188-DOE) ANTICIPATED TRANSIENTS WITHOUT SCRAM FOR LIGHT WATER REACTORS: IMPLICATIONS FOR LIQUID METAL FAST BREEDER REACTORS. Kastenber, W.E.; Solomon, K.A. (RAND Corp., Santa Monica, CA (USA)). Jul 1979. Contract ET-78-S-03-2243. 40p. Dep. NTIS, PC A03/MF A01.
In the design of light water reactors (LWRs), protection against anticipated transients (e.g., loss of normal electric power and control rod withdrawal) is provided by a highly reliable scram, or shutdown system. If this system should become inoperable, however, the transient could lead to a core meltdown. The Nuclear Regulatory Commission (NRC) has proposed, in NUREG-0460 [1], new requirements (or acceptance criteria) for anticipated transients without scram (ATWS) events and the manner in which they could be considered in the design and safety evaluation of LWRs. This note assesses the potential impact of the proposed LWR-ATWS criteria on the liquid metal fast breeder reactor (LMFBF) safety program as represented by the Clinch River Breeder Reactor Plant.
- 2761 (RAND-P--6226) REPROCESSING OF NUCLEAR FUEL AND PLUTONIUM BREEDER COMMERCIALIZATION: IMPLICATIONS OF DEFERRAL. Perry, W.D. (RAND Corp., Santa Monica, CA (USA)). Nov 1978. 64p. (CONF-781182--4). Rand Corp., Santa Monica, CA.

From Joint national ORSA/TIMS meeting; Los Angeles, CA, USA (13 Nov 1978).

U.S. government nuclear weapons anti-proliferation policies that pertain to the deferral of commercial-scale reprocessing of nuclear fuel and plutonium breeding reactors are reviewed. Prior U.S. economic cost/benefit analyses of spent nuclear fuel reprocessing and recycling are also summarized. The time and capacity of commercial-scale reprocessing facilities and the associated minimum present discounted costs that would support a later decision to proceed with U.S. breeder commercialization are analyzed.

- 2762 (RAND/R--2441-ACDA) COMPARISON OF THE CAPITAL COSTS OF LIGHT WATER REACTOR AND LIQUID METAL FAST BREEDER REACTOR POWER PLANTS. Mooz, W.E.; Siegel, S. (RAND Corp., Santa Monica, CA (USA)). Feb 1979. 71p. Rand Corp., Santa Monica, CA.

Recent studies by Atomics International, General Electric, and Westinghouse yielded capital cost estimates for a fifth of a kind LMFBR. After adjusting these estimates for plant size and converting them to 1978 dollars, they ranged from \$1043 to \$1495 per kWe, excluding interest during construction (IDC). These cost estimates were compared in this study with a LWR estimated by GE to cost \$900 per kWe, also excluding IDC. Projecting these costs far into the future entails great uncertainty. The principal factors that affect future capital cost ratios between the LMFBR and the LWR are: the initial cost ratio that is assumed for the 1977 design plants; assumptions about changes in scope resulting from the social, political, and regulatory climate; and differential construction times for the two types of plant. The cost ratios are relatively unaffected by the LMFBR deployment schedule and cost learning effects, differential scaling effects, and the LMFBR introduction date.

- 2763 (RD-B-N--4254) REPORT ON THE SECOND SPECIALISTS' MEETING ON REACTOR NOISE (SMORN II), GATLINBURG, TENNESSEE, SEPTEMBER 19TH-23RD 1977. Greef, C.P. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Mar 1978. 19p. Dep. NTIS (US Sales Only), PC A03/MF A01.

The current state of power reactor noise analysis as presented at the Second Specialists' Meeting on Reactor Noise (SMORN II) is reviewed. Practical successes include the analysis of PWR core support barrel motions and BWR instrument tube vibrations, leading to better understanding of the Physical Processes involved. Future applications to LMFBRs are being actively pursued, but more information is required, notably on background noise sources in operating reactors. Theoretical understanding is improving in a number of specific areas, but general fundamental progress is hampered by the multiplicity of potential noise sources in a reactor. Past attempts to automate noise analysis so that simplified information may be presented to the reactor operator have not been totally successful. This has resulted from an over-hasty application of the techniques before all of the factors involved in physical phenomena were fully appreciated.

- 2764 (RDT-E--6-19T(3-78)) RADIAL REFLECTOR ASSEMBLY FOR LIQUID METAL FAST BREEDER REACTORS. (Department of Energy, Washington, DC (USA). Div. of Nuclear Power Development). Mar 1978. Contract EY-76-C-14-2170. 13p. RSO.

This standard sets forth the requirements

for the fabrication, inspection, and quality assurance of core radial reflector assemblies for use in liquid metal fast breeder reactors.

- 2765 (RDT-E--6-33T(Amend.1)(6-78)) CONTROL ROD ASSEMBLY FOR LIQUID METAL FAST BREEDER REACTORS. (Department of Energy, Washington, DC (USA). Div. of Nuclear Power Development). 8 Sep 1978. Contract EY-76-C-14-2170. 14p. RSO.

This standard establishes the requirements for fabrication, testing, and inspection of control rod assemblies for use in liquid metal fast breeder reactors.

- 2766 (RDT-E--13-16T) DRIVER FUEL ASSEMBLY. (Department of Energy, Washington, DC (USA). Div. of Reactor Development and Demonstration). Sep 1977. Contract EY-76-C-14-2170. 12p. RSO.

The standard establishes the requirements for fabrication, testing, and inspection of fuel assemblies for use in liquid metal fast reactors.

- 2767 (RDT-F--3-40T) METHODS FOR THE ANALYSIS OF SODIUM AND COVER GAS. (Department of Energy, Washington, DC (USA). Office of Nuclear Energy Programs). Jan 1976. Contract EY-76-C-14-2170. 16p. RSO. LMFBR TYPE REACTORS; COOLANTS; COVER GAS; SODIUM; REACTOR COOLING SYSTEMS; QUALITATIVE CHEMICAL ANALYSIS; STANDARDS

- 2768 (RDT-INDEX--(01-79)) RDT STANDARDS INDEX. (Oak Ridge National Lab., TN (USA)). 19 Jan 1979. Contract W-7405-ENG-26. 35p. RSO.

The January 1979 Index of RDT Standards for the period October through December 1978 is presented. The attached 'New Publications List' identifies each standard, amendment, and/or revision that was published during this period. An order form is provided in the Index for your convenience in requesting those RDT standards that may be needed by your organization.

- 2769 (RE-A--78-241) AUTOMATED COMMON CAUSE ANALYSIS USING AUTOCC. Wilson, J.R. (EG and G Idaho, Inc., Idaho Falls (USA)). Oct 1978. Contract EY-76-C-07-1570. 32p. Dep. NTIS, PC A03/MF A01.

This is a user's manual for AUTOCC, a code which prepares common cause analysis input for COMCAN II. Basic events are automatically analyzed for common cause susceptibilities, and data about initiators is reformed into generic cause common locations. AUTOCC is written in FORTRAN IV for the CDC 7600.

- 2770 (RFP--2887) PYROCHEMICAL COPROCESSING OF UO_2 -- PUO_2 LMFBR FUEL BY THE SALT TRANSPORT METHOD. Knighton, J.B.; Baldwin, C.E. (Atomics International Div., Golden, CO (USA). Rocky Flats Plant). 1979. Contract EY-76-C-04-3533. 24p. (CONF-790415--29). Dep. NTIS, PC A02/MF A01.

From 177. national meeting of the American Chemical Society; Honolulu, HI, USA (1 Apr 1979).

The pyrochemical coprocessing of spent nuclear fuel by the Salt Transport Process appears to be a potentially viable reprocessing method, not only as an 'exportable proliferation resistant technology,' but as a domestic reprocessing operation. All operations are nonaqueous and waste generation is in solid

form, thus requiring no conversion from aqueous solutions to solids.

- 2771 (RHO-LD--66) ENGINEERING STUDY OF DRY NEAR-SURFACE INTERIM STORAGE OF FAST FLUX TEST FACILITY SPENT FUEL. (Atomics International Div., Richland, WA (USA)). Rockwell Hanford Operations). Nov 1978. Contract EY-77-C-06-1030. 66p. Dep. NTIS, PC A04/MF A01.

A brief study was conducted to assess methods for dry, near-surface storage of FFTF spent driver fuel assemblies in the 200 Area at Hanford. The disposable cask, drywell and vault concepts are the most promising approaches for FFTF spent fuel storage in the Hanford 200 Area starting in 1986. The cost and risk for the vault concept are higher than for either the disposable cask or drywell concepts. The disposal cask and drywell concepts are both satisfactory for this application. The risk for any of these concepts is minimal. Design and construction of a disposable cask or drywell facility can be completed within the constraint of a 1986 startup date if the project is initiated in FY-1980.

- 2772 (RISLEY-Trans--4009) EXTINGUISHMENT OF SODIUM FIRES. Malet, J.-C. (UKAEA Risley Nuclear Power Development Labs.). 1978. 8p. British Library Lending Division, Boston Spa.

The chemistry of sodium reactions with water and with oxygen is described, and the formation of toxic and opaque aerosols on ignition of sodium. Precautions taken against sodium fires in the PHENIX and SUPER-PHENIX reactors are discussed. A fire extinguisher powder called MARCALINA, containing alkaline carbonates and graphite, is described.

- 2773 (RLO--1830-T6) CONCEPTUAL DESIGN OF A SURVEILLANCE AND INSERVICE INSPECTION SYSTEM FOR THE PRIMARY PIPING AND COMPONENTS, FAST FLUX TEST FACILITY. PHASE C REPORT. Wylie, R.D.; Pickett, A.G.; Gerlach, C.R.; Hawkesworth, W. (Southwest Research Inst., San Antonio, TX (USA)). 16 Jan 1970. Contract EY-76-C-06-1830. 26p. Dep. NTIS, PC A03/MF A01.

The conceptual design of a Surveillance and Inservice Inspection (SISI) System for the primary coolant boundary outside of the reactor vessel area is presented. This portion of the system consists of the primary coolant piping between the reactor vessel area and the heat transport cells, and the piping, pumps, intermediate heat exchanger (IHX), valves and flowmeters in the cells. The SISI system for this part of FFTF and for the reactor vessel area will be integral, compatible sections of the complete FFTF-SISI system.

- 2774 (RLO--2227/T32-2) BURNING ACTINIDES IN VERY HARD SPECTRUM REACTORS. Robinson, A.H.; Shirley, G.W.; Prichard, A.W.; Trapp, T.J. (Oregon State Univ., Corvallis (USA)). Dept. of Nuclear Engineering). 20 Mar 1978. Contract EY-76-S-06-2227-032. 7p. (CONF-780622--73). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

The major unresolved problem in the nuclear industry is the ultimate disposition of the waste products of light water reactors. The study demonstrates the feasibility of designing a very hard spectrum actinide burner reactor (ABR). A 1100 MW/sub t/ ABR design fueled entirely with actinides reprocessed from light water reactor (LWR) wastes is proposed as both an ultimate disposal mechanism for actinides and a means of concurrently producing usable

power. Actinides from discharged ABR fuel are recycled to the ABR while fission products are routed to a permanent repository. As an integral part of a large energy park, each such ABR would dispose of the waste actinides from 2 LWRs.

- 2775 (RLO--2227/T32-6) STUDIES OF ACTINIDE WASTE BURNUP AND USE IN FAST SPECTRA. QUARTERLY REPORT, JANUARY 1--MARCH 31 1978. Robinson, A.H. (Oregon State Univ., Corvallis (USA)). Jul 1978. Contract EY-76-S-06-2227-032. 53p. Dep. NTIS, PC A04/MF A01.

Actinide Burning Reactors (ABR's) are assumed to consist of infinite arrays of actinides (i.e. no leakage), arranged in four different fuel compositions; each of which is in three different lattice arrangements. The consumption rate of the actinides is quite sensitive to the neutron energy spectrum, and the neutron flux level. The results obtained when the twelve different actinide configurations were placed inside an EBR-II driver region are presented. For these twelve cases the actinide region was 5cm in radius surrounded by a uranium fueled EBR-II driver region of 40cm radius. The core height was 40cm. Again a core average power density of 600kw/l was used. No reflector was included. Under these conditions the spectrum and flux levels are determined by the driver region. The maximum burnup of ^{241}Am was 32% while that for ^{243}Am was 20%. This lower burnup is due primarily to lower flux levels in the actinide region. The results show that approximately the same fractional burnup is obtained for all twelve cases. Thus, it appears essential that the driver region be optimized.

- 2776 (RRC--16) ANALYSIS OF THE BREEDING CAPABILITY OF VARIOUS FAST REACTOR FUELS. Reddy, C.P.; Lee, S.M.; Shankar Singh, R. (Reactor Research Centre, Kalpakkam (India)). Feb 1977. 19p. Dep. NTIS (US Sales Only), PC A02/MF A01.

A study of the breeding capability of a wide range of fast reactor fuels on recent evaluations of neutron cross sections is presented. A thin clad, sodium cooled, large fast reactor is considered fuelled with various combinations of the fissile materials Pu and ^{235}U and the fertile materials ^{238}U and Th. The fuel types studied are metal, oxide, carbide and nitride. The effect of variations in the structural steel fraction, the fuel smear density and the reactor excess reactivity for the plutonium uranium oxide fuel case is also presented. The results obtained are discussed in detail and give an indication of the expected breeding performance of other fast reactor fuels relative to that of the currently used plutonium-uranium oxide fuel.

- 2777 (RSA-TM--4) SURVEY OF THE CURRENT STATE OF KNOWLEDGE OF INCIPIENT BOILING SUPERHEAT IN SODIUM. Greer, B. (Mount Holyoke Coll., South Hadley, MA (USA)). 1979. Contract W-31-109-ENG-38. 38p. Dep. NTIS, PC A03/MF A01.

Superheat data obtained by various investigators indicate that many parameters affect this phenomenon. Controlling parameters appear to be inert gas concentration, oxide concentration, system pressure, pressure-temperature history, rate of temperature rise, heat flux, flow rate, operating time on the system, surface conditions, and radiation. Of these, the two believed most influential in controlling incipient boiling superheat are the inert gas concentration and oxide concentration. Experimental results for the

heat flux and rate of temperature rise appear to be the most inconsistent.

- 2778 (SAN--0115-097-T1) POTENTIAL HAZARD TO SECONDARY CONTAINMENT FROM HCDA-GENERATED MISSILES AND SODIUM FIRES. Romander, C.M. (SRI International, Menlo Park, CA (USA)). Feb 1979. Contract EY-76-C-03-0115-097. 32p. Dep. NTIS, PC A03/MF A01.

The potential hazard of HCDA-generated missiles is analyzed, and the current status of the potential hazards of sodium fires is summarized. Simple analyses are performed to determine lower bounds on the HCDA energetics required to generate missiles that could reach the secondary containment structure of a 1000-MWe LMFBR. The potential missiles considered include the vessel head, components mounted on the head, and control rods.

- 2779 (SAN--0115-T5) EXPERIMENTAL STUDIES IN CDA WORK-ENERGY ATTENUATION. QUARTERLY PROGRESS REPORT NO. 17, 1 JANUARY TO 31 MARCH 1979. Tobin, R.J.; Cagliostro, D.J.; Romander, C.M.; Goodale, T.C. (SRI International, Menlo Park, CA (USA)). 31 Mar 1979. Contract EY-76-C-03-0115. 22p. Dep. NTIS, PC A02/MF A01.

The objectives of this program are to: develop a basic understanding of the dynamics and thermodynamics of expanding bubbles similar to the core disassembly accident (CDA) bubble in liquid metal fast breeder reactors (LMFBR); determine experimentally the structural response of an LMFBR to simulated CDA loads resulting from a postulated fuel vapor expansion; investigate potential hazards of CDA-generated missiles and sodium fires to the secondary containment system of the reactor; and investigate phenomena that may limit the work potential of HCDA bubble expansions.

- 2780 (SAN--0115-T6) EXPERIMENTAL STUDIES IN CDA WORK-ENERGY ATTENUATION. QUARTERLY PROGRESS REPORT NO. 18, 1 APRIL-30 JUNE 1979. Tobin, R.J.; Cagliostro, D.J.; Romander, C.M. (SRI International, Menlo Park, CA (USA)). 30 Jun 1979. Contract EY-76-C-03-0115-097. 17p. Dep. NTIS, PC A02/MF A01.

A study on the CDA energetics necessary to generate missiles that could reach the secondary containment of a large-LMFBR is described. Scaling and prototypicality analyses on CDA work-energy attenuation are also described.

- 2781 (SAN--0115-T7) NONEQUILIBRIUM PHENOMENA IN FLASHING WATER USED AS AN HCDA SOURCE SIMULANT. TECHNICAL REPORT NO. 8. Tobin, R.J.; Cagliostro, D.J.; Florence, A.L. (SRI International, Menlo Park, CA (USA)). Poulter Lab.). Nov 1979. Contract EY-76-C-03-0115-097. 118p. Dep. NTIS, PC A06/MF A01.

The philosophy behind the design of liquid metal fast breeder reactors (LMFBR) is to keep the risk to the public from accidental releases of radioactivity extremely small. Consistent with this philosophy, LMFBR safety technology research sponsored by DOE includes the consideration of hypothetical core disruptive accidents (HCDA). Recent scale-model simulation experiments on HCDAs identified nonequilibrium phenomena in flashing liquids as a possible attenuation mechanism in the energetics of flashing reactor fuels during an HCDA.

- 2782 (SAN--0115-T8) STRUCTURAL RESPONSE OF 1/20 SCALE MODELS OF ABOVE CORE STRUCTURES TO

INCREASINGLY ENERGETIC HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS. TECHNICAL REPORT 7. Romander, C.M. (SRI International, Menlo Park, CA (USA)). Jul 1979. Contract EY-76-C-03-0115-097. 103p. Dep. NTIS, PC A06/MF A01.

Six experiments were performed to evaluate the structural response of an above core structure (ACS) to increasingly energetic hypothetical core disruptive accidents (HCDA). The tests were performed in a thickwall 1/20-scale model of a liquid metal fast breeder reactor (LMFBR) demonstration plant to provide information on ACS displacement as a function of HCDA energy so that computer codes such as SIMMER might include ACS displacement when calculating core release energetics. The 1/20-scale ACS models include a 5-kg aluminum block with dimensions that simulate the overall geometry of the ACS and the four columns that support the ACS over the core of the reactor. Each column includes an 8-in-long, 0.7-in-diameter 0.050-in-wall Ni 200 tubular section that experiences all of the deformation upon HCDA loading. The columns are supported by a rigid structure that simulates the geometrical constraints provided by the cover of the reactor.

- 2783 (SAN--0824-027-T1) HIGH QUALITY TUBE-TO-TUBESHEET JOINTS - WHEN, WHERE AND WHY. Lillie, A.F. (Rockwell International Corp., Canoga Park, CA (USA). Energy Systems Group). [nd]. Contract EY-76-C-03-0824-027. 12p. Dep. NTIS, PC A02/MF A01.

An autogeneous butt weld using the internal bore weld process is described. The weld is designed for the CRBR steam generator tubes to tubesheets which are fabricated of 2 1/2 Cr-1 Mo low alloy ferritic steel. As a precaution against in-service weld cracking, each weld is heat treated prior to inspection. (FS)

- 2784 (SAN--0824-T1) SODIUM LEAK DETECTION VERIFICATION TEST RESULTS. Witherspoon, J.E. (Rockwell International Corp., Canoga Park, CA (USA)). 27 Jul 1979. Contract EY-76-C-03-0824-012. 230p. AT.

Portions of document are illegible.

The results are presented for Sodium Leak Detection Verification Tests. In these tests, the required performance of selected CRBRP sodium-to-gas leak detection methods, which have application in LMFBR plants, was verified. Testing, in accordance with N707DRS520001, was performed by ETEC personnel. Copies of ETEC reports on the tests are included in Appendix A which comprises Volume 2 of this report.

- 2785 (SAN--0824-T2) SODIUM LEAK DETECTION, LONG TERM PERFORMANCE TEST: RESULTS FOR FY 1978. Witherspoon, J.E. (Rockwell International Corp., Canoga Park, CA (USA)). 20 Jun 1979. Contract EY-76-C-03-0824-012. 36p. AT.

The results are presented for the first 18 months of the long-term performance tests of aerosol-type sodium leak detectors. The equipment, which was designed and assembled at AI-ESG, is installed at EBR-II, and the operation is monitored by EBR-II personnel.

- 2786 (SAN--0824-T3) NATURAL CONVECTION: SODIUM LEAK DETECTION TESTS. Amaro, G.; Morejon, J. (Rockwell International Corp., Canoga Park, CA (USA)). 15 Jul 1976. Contract EY-76-C-03-0824-012. 103p. AT.

Aerosol type leak detectors were tested to obtain information regarding their response time to sodium leaks when monitoring the test

cell gas alone. Tests were conducted using both 24-in. and 3-in. diameter pipes. The test results showed that effective detection of small sodium leaks in vented 24-in. diameter pipes (2 gm/H) by aerosol detectors can be accomplished at temperatures greater than 600°F. At lower temperatures aerosol detectors are less effective.

- 2787 (SAN--0824-T4) SODIUM LEAK DETECTION LONG TERM PERFORMANCE TEST: RESULTS FOR FIRST SIX MONTHS. Witherspoon, J.E. (Rockwell International Corp., Canoga Park, CA (USA)). 5 Jun 1978. Contract EY-76-C-03-0824-012. 34p. AT.

The results are described for the first 6 months of the long-term performance tests of aerosol-type sodium leak detectors. The operation of the equipment, installed at EBR-II, is monitored by EBR-II personnel.

- 2788 (SAN--0824-T5) LEAK DETECTION RESPONSE TIME CHARACTERIZATION AND OPTIMIZATION TEST ANALYSIS. Chang, S.; Morejon, J.; Griffin, C.W. (Atomics International Div., Canoga Park, CA (USA)). 20 Oct 1975. Contract EY-76-C-03-0824-012. 53p. AT.

A general sodium-to-gas leak detector evaluation, based on the test condition effects on the aerosol production and the test parameter effects on the differential pressure (DP) detectors sensitivity is discussed. Recommendations are made to further improve the sensitivity and response time of the DP detector which proved to have a high potential in sodium leak detection, and for optimum use of other candidate leak detection devices. Cable detectors, hydrogen detectors, and chemical analysis of filter pads were seen to have high potential for leak detection in special environmental conditions or critical locations.

- 2789 (SAN--0824-T6) SMALL PIPE - NATURAL CONVECTION - SODIUM LEAK DETECTION TESTS. Morejon, J.A. (Rockwell International Corp., Canoga Park, CA (USA)). 2 Dec 1976. Contract EY-76-C-03-0824-012. 13p. AT.

Sodium leaks were simulated in a small pipe (3 in. diameter) at various temperatures and environmental conditions. The small pipe was provided with a uniform annular space of approximately 1/2 inch in diameter. The results from these tests when compared with previous tests conducted with a small pipe with no annular space indicate that the annulus plays an important part in enabling rapid detection of sodium leaks.

- 2790 (SAN--0824-T7) SODIUM LEAK DETECTION VERIFICATION TEST RESULTS. Witherspoon, J.E. (Rockwell International Corp., Canoga Park, CA (USA)). 27 Jul 1979. Contract EY-76-C-03-0824-012. 54p. AT.

This report describes the results of Sodium Leak Detection Verification Tests. In these tests, the required performance of selected CRBRP sodium-to-gas leak detection methods, which have application in LMFBR plants, was verified. Testing, in accordance with N707DR520001, was performed by ETEC personnel. Copies of ETEC reports on the tests are included in Appendix A which comprises Volume 2 of this report.

- 2791 (SAN--0824-T8) SODIUM LEAK DETECTION. Fletcher, F.; Gajewski, W.M. (Rockwell International Corp., Canoga Park, CA (USA)). [nd]. Contract EY-76-C-03-0824-012. 102p.

AT.

Research and development in sodium leak detection is reviewed. Topics covered include leak detector types, applications and requirements, development test programs, and analytical efforts and systems sensitivities.

- 2792 (SAN--0824-T9) SODIUM-TO-GAS LEAK DETECTION MOCKUP TESTS. Morejon, J.A. (Atomics International Div., Canoga Park, CA (USA)). 23 Oct 1975. Contract EY-76-C-03-0824-012. 66p. AT.

Several types of sodium leak detectors were tested using a pipe configuration and environmental conditions typical of LMFBR's. Aerosol/vapor monitors as well as electrical type monitors were tested. Reliable detection of small sodium leaks in the upper sodium temperature range (600 to 1000°F) was observed for all instrumentation except the contact (spark plug) detector. The gas chromatograph used for hydrogen detection gave definite indication over this temperature range as well as the lower sodium temperature range (400 to 600°F).

- 2793 (SAN--0824-T10) SUMMARY OF DESIGN AND DEVELOPMENT STATUS OF THE LIQUID METAL TO GAS LEAK DETECTION SYSTEM FOR THE CRBRP. Morejon, J.A. (Rockwell International Corp., Canoga Park, CA (USA)). 20 Aug 1976. Contract EY-76-C-03-0824-012. 70p. AT.

This TI updates the material presented in TI-N099TI661006 which discussed the Reference Liquid Metal to Gas Leak Detection System capabilities, performance, preliminary design requirements, safety aspects, and costs. A series of meetings took place with the various RM's and with NRC to discuss the system design and the leak detection requirements. Background information as well as the leak detection requirements are presented in this TI.

- 2794 (SAN--0824-T11) INVESTIGATION AND RATING OF SODIUM-TO-GAS LEAK DETECTION TECHNIQUES. Nelson, C. (Atomics International Div., Canoga Park, CA (USA)). 6 Feb 1973. Contract EY-76-C-03-0824-012. 30p. AT.

The investigation and rating of the sodium-to-gas leak detection techniques are presented based upon the leak detection requirements for the LMFBR Demonstration plant. The results of the investigation indicate that a variety of detection techniques are required for complete surveillance of the plant.

- 2795 (SAN--0824-T12) TEST REPORT: SODIUM-TO-GAS LEAK DETECTION SYSTEM SCREENING TESTS. Nelson, C.T. (Atomics International Div., Canoga Park, CA (USA)). 20 May 1975. Contract EY-76-C-03-0824-012. 17p. AT.

The data indicate that the filter plugging sensor, when chemical analysis is included, will meet the leak sensitivity and time response requirements for both primary and secondary heat transfer systems in an LMFBR. The cable detectors are useful only under special conditions. It is recommended that the aerosol tests be run for sufficiently long time periods to demonstrate that a direct signal (i.e., a significant pressure drop across the filter) could be reliably obtained.

- 2796 (SAN--0824-T13) SODIUM-TO-GAS LEAK DETECTION SYSTEM PERFORMANCE EVALUATION. Morejon, J.A. (Atomics International Div., Canoga Park, CA (USA)). 26 Jul 1976. Contract EY-76-C-03-0824-012. 99p. AT.

The Sodium to Gas Leak Detection System described in SDD 66, Rev 0, was evaluated for performance and capability to meet the Leak Detection requirements recently established by the various RM's (Reactor Manufacturers) for their respective systems. Results of this study are presented in this document, along with a revised cost estimate of the Reference System.

- 2797 (SAN--0824-T14) SELECTION OF AEROSOL DETECTORS FOR CRBRP. Morejon, J.A. (Atomics International Div., Canoga Park, CA (USA)). 29 Oct 1976. Contract EY-76-C-03-0824-012. 14p. AT.

Several types of Liquid Metal Aerosol to Gas Leak Detectors were tested at AI and LMEC to determine their sensitivity and performance in detection of sodium leaks. This information coupled with cost information was utilized to recommend specific types of aerosol detectors for CRBRP applications.

- 2798 (SAN--0893-T1) PROGRAM PLAN: ACOUSTIC LEAK DETECTION/LOCATION DEVELOPMENT AT GE-ARSD. (General Electric Co., Sunnyvale, CA (USA)). Advanced Reactor Systems Dept. [nd]. Contract EY-76-C-03-0893-031. 9p. Dep. NTIS, PC A02/MF A01.

Provide the development and subsequent specification, design and testing of an acoustic leak protection system which will detect a leak within a LMFBR steam generator. The goal for this system is to be at least as rapid and no more expensive than the chemical leak detection system under development for the Clinch River Breeder Reactor Plant (CRBRP).

- 2799 (SAN--1011-113R) CORRELATION OF GROUND RESPONSE SPECTRA WITH MODIFIED MERCALLI SITE INTENSITY. TOPICAL REPORT. Werner, S.D.; Ts'ao, H.S. (Agabian Associates, El Segundo, CA (USA)). 30 Sep 1978. Contract EY-76-C-03-1011. 168p. AT.

The objective of this study was to develop and update seismic design criteria and analysis methods for future LMFBR facilities. This report considers one aspect of this overall objective, namely, the development of intensity-dependent spectra for the following purposes: (1) to investigate ground response characteristics; (2) to provide a basis for assessing results from current procedures for defining seismic input criteria; and (3) to form the basis for new recommendations for developing criteria spectra to represent a given site intensity.

- 2800 (SAN--1011-114) STUDY OF VERTICAL GROUND RESPONSE SPECTRUM SHAPES. Werner, S.D.; Ts'ao, H.S. (Agabian Associates, El Segundo, CA (USA)). Aug 1977. Contract EY-76-C-03-1011. 71p. AT.

The report provides the results of an investigation of the effect of different normalizing procedures on the shape of response spectra for vertical earthquake ground motion. Vertical free-field earthquake ground motion records processed at the California Institute of Technology were used for the study. Spectra shape comparisons were made for composite spectra (mean and mean $\pm$ sigma) derived from spectra normalized to peak ground acceleration, peak ground velocity, and peak ground displacement, respectively. Spectra shapes were also derived by a three segment method and from natural (i.e., not normalized) spectra. Effects of local geology and local soil conditions on response spectra shapes were also investigated. Vertical spectra were developed for five different damping ratios using the three

segment method. The results of the investigation indicate that spectra developed from the statistical analysis of natural ground motion data has two advantages over spectra developed from the statistical analysis of normalized spectra shapes, namely (1) the spectra will not be distorted by the normalizing process; and (2) the probability of exceedance of the composite spectra can be well defined rather than approximated. Comparisons of composite spectra developed from both natural data and from normalized spectra with Regulatory Guide 1.60 procedures indicate that Regulatory Guide 1.60 procedures lead to vertical ground response spectra that are overly conservative. Comparisons of spectra for different local site conditions indicate that the method selected for normalizing the data has an important bearing on the extent to which local site effects appear to influence the spectrum shape.

- 2801 (SAN--1011-116) EVALUATION OF SEISMIC ANALYSIS TECHNIQUES FOR LMFBR PIPING AND EQUIPMENT. (Agabian Associates, El Segundo, CA (USA)). Jan 1978. Contract EY-76-C-03-1011. 188p. AT.

The development of efficient and reliable methods for calculating the seismic forces on piping and equipment critical to nuclear power plant safety has been the subject of numerous studies. In an effort to evaluate the procedures currently used for deriving piping and equipment design spectra, the report (a) discusses considerations that arise in the aseismic design of equipment/structure systems, such as characterization of earthquake inputs and modeling of equipment/structure systems; (b) assesses calculation techniques of in-structure response spectra; (c) evaluates a currently used approximate seismic analysis method; and (d) presents an "exact" probabilistic analysis of a coupled equipment/structure system under earthquake-like excitation. The study concludes that of the three methods evaluated and compared for accuracy, feasibility, and confidence level of conservatism in the results of the analysis, the probabilistic approach is the most rational, the time-history method is the next, and the "direct methods" are the least rational. Additional work is required to reduce the probabilistic method to a simple, practical design procedure.

- 2802 (SAN--1011-117) SEISMIC DESIGN VERIFICATION OF LMFBR STRUCTURES. (Agabian Associates, El Segundo, CA (USA)). 1 Jul 1977. Contract EY-76-C-03-1011. 119p. TIC \$6.50.

The report provides an assessment of the seismic design verification procedures currently used for nuclear power plant structures, a comparison of dynamic test methods available, and conclusions and recommendations for future LMFBR structures.

- 2803 (SAN--1011-118R) ASSESSMENT OF SEISMIC ANALYSIS PROCEDURES FOR LMFBR IN-CORE COMPONENTS. (Agabian Associates, El Segundo, CA (USA)). Mar 1979. Contract EY-76-C-03-1011. 71p. Dep. NTIS (US Sales Only), PC A04/MF A01.

The report provides an assessment of state-of-the-art dynamic seismic analysis procedures for in-core components of liquid metal fast breeder reactors. It is based on a review of both United States and foreign published literature and upon discussions with technical personnel presently engaged in performing seismic design analyses of reactor in-core

components.

- 2804 (SAN--1011-200) TEST VERIFICATION OF SEISMIC RESPONSE OF LMFBR PIPING SYSTEMS. QUARTERLY TECHNICAL PROGRESS REPORT. (Agbabian Associates, El Segundo, CA (USA)). Jan 1979. Contract EY-76-C-03-1011. 96p. AT.

The objective of this investigation is to experimentally verify the analytically predicted dynamic response of a typical breeder reactor piping system to the design seismic input motions. This is a progress report on the first phase of a two-phase investigation. The objective of the first phase is to develop a test plan which would include the selection of the system to be tested, definition of the analytical and experimental studies to be performed, and the preparation of a cost estimate for the second phase. The second phase of the investigation is to provide the tests and supporting analyses. This report summarizes work performed through October 1978. Work performed consists of a background review, definition of a typical breeder reactor piping system and studies to define the problems associated with both the analytical and experimental portions of the verification program.

- 2805 (SAN--1011-202) EVALUATION OF SEISMIC ANALYSIS TECHNIQUES FOR BREEDER REACTOR PIPING AND EQUIPMENT. QUARTERLY TECHNICAL PROGRESS REPORT. (Agbabian Associates, El Segundo, CA (USA)). Apr 1979. Contract EY-76-C-03-1011. 116p. AT.

Progress is reported on an investigation being conducted to evaluate different methods used to predict the response of nuclear power plant piping and equipment subsystems to earthquake vibratory ground motions. The program is primarily concerned with simplified methods based on free-field response spectra as input rather than on time-history methods of analysis. The report summarizes results of investigations conducted since the issue of Topical Report SAN/1011-116 which provided an evaluation of an approximate direct amplification method based on a comparison of results obtained by this method with exact solutions based on four time-history input records. This study has been extended to provide a probabilistic evaluation based on ten time-history analyses. Approximate methods of seismic analysis of multisupported piping and equipment based on input spectra methods is also under investigation. Proximity spectra for two oscillators subjected to propagating nonstationary random excitation have been developed and are included in the report.

- 2806 (SAN--1359-T1) MATHEMATICS CLINIC. NUCLEAR SAFETY CODE STUDY. (Harvey Mudd Coll., Claremont, CA (USA)). Jan 1980. Contract EG-77-S-03-1359. 31p. Dep. NTIS, PC A03/MF A01.

What would happen during an overpower accident in an LMFBR is analyzed. To assist studying the problem, Rockwell Energy Systems Group is currently developing a model of an overpower accident. It is the responsibility of this year's Nuclear Safety Code Study team to develop a useable algorithm for numerical evaluation of the stiff equations which are expected to constitute a majority of the model.

- 2807 (SAN/0700--1) APPLICATIONS OF CAVITATION ANALYSIS TO DESIGN AND OPERATION OF SPTF-HFL PUMP LOOPS. Bowers, W.M. (Liquid Metal Engineering Center, Canoga Park, CA

(USA)). 1977. Contract EY-76-C-03-0700. 35p. AT.

A consistent set of cavitation equations has been developed in a form suitable for conceptual design of SPTF (Sodium Pump Test Facility) modifications or use as an operational aid. However, to apply the equation as design tools or operational aids, certain cavitation test data must be available. It is recommended that incipient cavitation conditions be induced on valve HV=130 in the existing loop and the acoustic data on cavitation be collected.

- 2808 (SAN/1112--1) APRICOT PROGRAM: COMPARISON AND BENCHMARKING OF COMPUTATIONAL METHODS FOR ANALYSIS OF LMFBR STRUCTURAL RESPONSE TO POSTULATED CORE DISRUPTIVE ACCIDENTS. PHASE I REPORT. (Science Applications, Inc., Oakland, CA (USA)). Oct 1977. Contract EY-76-C-03-1112. 103p. Dep. NTIS, PC A06/MF A01.

The results are presented of Phase I of the APRICOT (Analysis of Primary Containment Transients) Program. APRICOT is an international cooperative activity for comparison and benchmarking of computational methods used to analyze LMFBR structural response to pressure loads from CDA's (Core Disruptive Accidents). Phase I involved parallel calculations of three computer problems by scientific groups in Europe, Japan and the United States. Independent experts reviewed these calculations for the purpose of comparing computational results and methods of solution. The main conclusion was that all calculations were in reasonable agreement, although there were some basic differences in details and approach to solution of the problems.

- 2809 (SAND--76-0614) FLOW VISUALIZATION IN HEAT-GENERATING POROUS MEDIA. Lee, D.O.; Nilson, R.H. (Sandia Labs., Albuquerque, NM (USA)). Nov 1977. Contract EY-76-C-04-0789. 33p. Dep. NTIS, PC A03/MF A01.

The work reported is in support of the Sandia Post-Accident Heat Removal Program, in which simulated LMFBR beds will be subjected to in-pile heating in the ACPR (Annular Core Pulsed Reactor). Flow visualization experiments were performed to gain some insight into the flow patterns and temperature distributions in a fluid-saturated heat-generating porous medium. Although much of the information presented is of a qualitative nature, it is useful in the recognition of the controlling transport process and in the formulation of analytic and numerical models.

- 2810 (SAND--77-0161C) MOBILE HELIUM COOLING LOOP SYSTEM FOR IN-REACTOR PAHR EXPERIMENTS. Brammer, J.A. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 8p. (CONF-771109--22). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

One objective of the PAHR program at Sandia Laboratories is to better simulate the decay heating of debris beds and molten pools by using the Annular Core Pulse Reactor (ACPR) operated in its steady-state mode to fission heat UD_2 fueled PAHR experiment capsules. Safe performance of these typically long duration, elevated temperature PAHR experiments within the ACPR first necessitated the development of a unique system for cooling, control, and diagnostics of the PAHR experiment capsule. The experimental equipment arrangement of the mobile helium cooling loop (HCL) system

connected through piping to the PAHR debris bed capsule located at the bottom of the 23 cm diameter by 7 m long dry experiment tube penetrating the core region of the pool-type ACPR reactor is described. The mobile HCL system uses low pressure helium gas to remove heat at rates up to 20 kW that was fission generated within the sealed capsule containing UO_2 and molten sodium.

- 2811 (SAND--77-0611) FAST REACTOR SAFETY RESEARCH PROGRAM. QUARTERLY REPORT, JANUARY--MARCH 1977. (Sandia Labs., Albuquerque, NM (USA)). Aug 1977. Contract EY-76-C-04-0789. 144p. (NUREG--0181-2). Dep. NTIS, PC A07/MF A01.

Progress is summarized in the following study areas: accident energetics; core debris behavior; sodium containment and structural integrity; research for elevated temperature design criteria; fuel motion detection; and ACPR fuel motion detection system.

- 2812 (SAND--77-0723) FIRE PROTECTION DEVICE FOR LMFBR SPENT FUEL SHIPPING CASKS. Moss, M.; Heckman, R.C. (Sandia Labs., Albuquerque, NM (USA)). Aug 1977. Contract EY-76-C-04-0789. 21p. Dep. NTIS, PC A02/MF A01.

This report describes the analysis and experimental evaluation of a device to provide fire protection for a shipping cask used to transport liquid metal fast breeder reactor (LMFBR) fuel rods. Thermal analyses of various fire protection schemes were conducted by means of the finite difference code, CINDA. The choice of materials for the cask body was determined to be less important than the dimensions and number of cooling fins attached to the cask. Of several protection methods considered, radiation/convection shields between the cooling fins looked most attractive; these were tested on a quarter-scale cask model. The shields consisted of bimetallic bands which expanded when heated, blocking the fire, and retracted when cooled. This automatic reversibility would allow the fins to dissipate internally generated heat of a full-size cask once the fire was out. Tests showed that, even in an asymmetric fire exposure, the bands expanded to provide protection to the cask. The directly exposed surface of the cask model, protected this way, reached a temperature of only 180°C compared with 295°C in the unprotected state, when subjected to a butane flame for 0.5 h at a distance of 15 cm. Greater relative effectiveness could be expected in an engulfing fire in which the bands would expand more symmetrically.

- 2813 (SAND--77-0924C) EVALUATION OF COMPUTATIONAL TECHNIQUES FOR LMFBR SAFETY ANALYSIS. Herrmann, W.; Gross, M. (Sandia Labs., Albuquerque, NM (USA); Science Applications, Inc., Oakland, CA (USA)). 1977. Contract EY-76-C-04-0789. 40p. (CONF-780803-1). Dep. NTIS, PC A03/MF A01.

From 4. conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1978).

Analysis of Primary Containment Transients (APRICOT) is an ERDA sponsored project in which a variety of reactor safety analysis groups around the world have been invited to participate by performing calculations to verify capabilities of large computer codes used to analyze postulated core disruptive accidents of liquid metal fast breeder reactors. Nine groups have performed calculations of the first three problems which were set, using ten computer codes. Two problems were simple test problems for which

analytical solutions exist, namely an ideal gas shock tube, and a suddenly pressurized spherical cavity in an infinite elastic medium. The third problem concerns an explosion in a partially water-filled overstrong cylindrical containment vessel for which experimental data exist. A critique of the results of these calculations is given in the paper.

- 2814 (SAND--77-0971C) RADIATION SHIELD DESIGN FOR LMFBR SPENT-FUEL SHIPPING CASKS. Dupree, S.A. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 8p. (CONF-771109--68). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

Previous analyses have examined a limited number of the alternatives available for designing an LMFBR spent-fuel shipping cask (SFSC) using a non-volatile neutron shield; i.e., a neutron shield which will not be lost in an accident involving a fire. The present study extends the scope of these hypothetical designs to include combinations of volatile and non-volatile neutron shield materials.

- 2815 (SAND--77-1014C) CODED APERTURE IMAGING OF REACTOR ILLUMINATED LMFBR FUEL PINS. Kelly, J.G.; Stalker, K.T. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 5p. (CONF-771109--37). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

As a part of Sandia Laboratories program to study simulated core disruptive accidents in reactor safety research, a fuel motion detection system based on coded aperture imaging is being developed. Until the experiments reported here were conducted it was not known whether either of two important questions could be properly answered. First, could an actual fuel pin illuminated by a reactor be imaged with a coded aperture and an active detector system with acceptable frame rates and spatial resolution. Second, could collimator and shield structures be fabricated for the Annular Core Pulsed Reactor (ACPR), for which the system is being built, that would provide acceptable signal to background ratios. Both of these questions can now be answered in the affirmative because of the favorable experiments recently conducted at the Sandia Pulsed Reactor (SPR II) and at the ACPR.

- 2816 (SAND--77-1015C) IN-CORE FUEL MOTION DETECTION IN THE SANDIA PULSE REACTOR. Wright, S.A.; Renken, J.H.; Kroeger, F.R. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 6p. (CONF-771109--5). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

One of the major fuel motion diagnostic techniques under consideration for use in large scale fast reactor safety tests is in-core fuel motion detection. The Nuclear Regulatory Commission is funding a program at Sandia Laboratories to experimentally test the feasibility of using in-core neutron detectors to monitor fuel motion in medium- to large-scale tests. The new areas of research required by this program are the development of miniature neutron detectors which can be placed in hollow wire wrap and the development of unfolding methods which relate detector response to fuel motion.

- 2817 (SAND--77-1016C) HYDRODYNAMIC ANALYSIS

OF THE LMFBR PROMPT BURST EXCURSION (PBE) EXPERIMENT. Young, M.F. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 7p. (CONF-771109--7). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

A series of in-pile experiments has been conducted at Sandia Laboratories to provide information on pressure levels and conversion of thermal energy into mechanical work in LMFBR cores during hypothetical, superprompt-critical excursions. Pressures generated in these experiments are recorded by a pressure transducer located at the top and bottom of a sodium channel surrounding a single, fresh UO_2 fuel pin. Work energy conversion is measured by a linear motion transducer connected to a piston at the top of the sodium column. Since the pressure transducers are located fairly far from the location of pin failure, it becomes necessary to determine the effect of channel geometry and piston motion on the observed pressure data. A two-dimensional, hydrodynamic analysis of pressure pulse propagation in the fuel pin-coolant channel geometry was therefore performed using the CSQII computer code. The initial series of PBE experiments consists of single, fresh UO_2 pins surrounded by a sodium-filled or dry-coolant channel contained in a closed test capsule. The capsule is subjected to a maximum pulse in the Annular Core Pulse Reactor (ACPR) resulting in an energy deposition of from 2350 to 2900 J/g (14 and 20 percent enriched pins). The pulse width at half maximum (PWHM) is about 5 ms.

2818 (SAND--77-1017C) EFFECT OF THERMAL STRESS ON OXIDE CRUSTS. Ostensen, R.W. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 4p. (CONF-771109--9). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

In the transition phase in LMFBR hypothetical accidents, the existence of an insulating crust of frozen fuel between molten fuel and cold structure is of key importance. The disruption of such a crust for fuel flowing through the axial blanket regions can lead to rapid freezing and subsequent blockage formation. In a boiling pool of molten fuel and steel, the presence of the crust may be necessary in order to reduce the convective heat transfer rate and allow a subcritical boilup. This paper describes the effect of thermal stress on such a crust and points out that such crusts may be far weaker than previously believed.

2819 (SAND--77-1018C) FISSION-GAS BUBBLE MODELING FOR LMFBR ACCIDENTS. Ostensen, R.W. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 10p. (CONF-771109--12). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

The behavior of fission-gas bubbles in unrestructured oxide fuel can have a dominant effect on the course of a core disruptive accident in an LMFBR. The paper describes a simplified model of bubble behavior and presents results of that model in analyzing the relevant physical assumptions and predicting gas behavior in molten fuel.

2820 (SAND--77-1032C) RATE EFFECTS IN THE DYNAMIC VAPORIZATION OF URANIUM DIOXIDE. Benson, D.A. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 6p. (CONF-771109--23). Dep. NTIS, PC A02/MF

A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

Information for vaporizing UO_2 at temperatures and vaporization rates appropriate to fast reactor safety analysis is now becoming available from several different types of experiments. These data are summarized and compared to models used in reactor safety analysis. It has been commonly assumed in fast reactor analysis that the vaporization of fuels at high temperatures can be described by an equation of state for material in chemical and thermal equilibrium. However, the comparisons of dynamic experiments in oxide material to equilibrium measurements produce contradicting results. These observations can be resolved by a rate dependent effect in the high temperature oxide.

2821 (SAND--77-1059C) LARGE SCALE SODIUM INTERACTIONS. PART 2. PRELIMINARY TEST RESULTS FOR LIMESTONE CONCRETE. Smaardyk, J.E.; Sutherland, H.J.; King, D.L.; Dahlgren, D.A. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 4p. (CONF-771109--47). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

Any sodium cooled reactor system must consider the interaction of hot sodium with cell liners, and given either a failed liner or a hypothetical core disruptive accident, the interaction of hot sodium with concrete. The data base available for safety assessments involving these interactions is limited, especially for the concrete and failed liner interactions. To better understand what happens when hot sodium comes in contact with concrete, a series of tests is being carried out to investigate sodium-concrete reactions under conditions which are similar to actual reactor accident conditions. Tests cover the cases of sodium spills on bare concrete and on cells with defective steel liners. Specific objectives have been to obtain a complete description of the sodium/concrete interaction including heat balance, gas evolution and flow, movement and heat generation of the reaction zone, reaction product formation, and the layering or movement of the products.

2822 (SAND--77-1073C) LARGE SCALE SODIUM INTERACTIONS. PART 1. TEST FACILITY DESIGN. King, D.L.; Smaardyk, J.E.; Sallach, R.A. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 7p. (CONF-771109--46). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

During the design of the test facility for large scale sodium interaction testing, an attempt was made to keep the system as simple and yet versatile as possible; therefore, a once through design was employed as opposed to any type of conventional sodium 'loop.' The initial series of tests conducted at the facility call for rapidly dropping from 20 kg to 225 kg of sodium at temperatures from 825°K to 1125°K into concrete crucibles. The basic system layout is described. A commercial drum heater is used to melt the sodium which is in 55 gallon drums and then a slight argon pressurization is used to force the liquid sodium through a metallic filter and into a dump tank. Then the sodium dump tank is heated to the desired temperature. A diaphragm is mechanically ruptured and the sodium is dumped into a crucible that is housed inside a large steel test chamber.

2823 (SAND--77-1092C) NETWORK MODEL OF FREE CONVECTION WITHIN INTERNALLY HEATED POROUS MEDIA. Conrad, P.W. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 6p. (CONF-771109--15). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

A hypothetical core-disruptive accident (HCDA) in a liquid metal fast breeder reactor (LMFBR) may result in the formation of an internally heated debris bed. Considerable attention has been given to postulated mechanisms by which such beds may be cooled. It is the purpose of the work described to demonstrate a method for computing the heat transfer from such a bed to the overlying sodium pool due to single-phase, free convection.

2824 (SAND--77-1093C) THERMAL BEHAVIOR OF CONCRETE. Knight, R.L. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 7p. (CONF-771109--18). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

The treatment of post accident containment problems requires an understanding of concrete when subjected to high temperatures due to thermal contact with molten core materials and/or coolant. Sodium spills on concrete are of special concern because the resulting chemical reactions are collectively exothermic. Among other possible safety problems, the gas released by the thermal decomposition of concrete has the potential of producing pressures sufficient to endanger the integrity of cell liners. Insight into these phenomena has been gained by constructing a computer model and performing a series of studies. The code presently calculates the following quantities as functions of space and time: (1) temperature of the cell liner, (2) temperature of the concrete, (3) ablation rate of the heated concrete surface, (4) the water, water vapor, and carbon dioxide evolution rates, (5) the total gas discharged, and (6) the decomposition energy absorption rate for each decomposing species. The code has been used to predict the decomposition of concrete both with and without a containment liner.

2825 (SAND--77-1094C) EFFECTIVE EQUATION-OF-STATE MEASUREMENTS ON URANIUM DIOXIDE. Reil, K.O.; Cronenberg, A.W. (Idaho National Engineering Lab., Idaho Falls (USA)). 1977. Contract EY-76-C-04-0789. 7p. (CONF-771109--4). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

The safety analysis of Liquid Metal Fast Breeder Reactor (LMFBR) systems requires knowledge of the vapor pressure of reactor fuels up to 6000 K. A new technique was developed to obtain experimental vapor pressure data for reactor fuel materials at high temperatures and pressures. Fuel samples are pulse fission heated in the Annular Core Pulse Reactor (ACPR) at heating rates comparable to those anticipated during a severe LMFBR transient. Since the vapor of reactor fuels may contain several dissociated fuel species, volatile fission products, and/or fission gas, and the evolution of that pressure may be time or rate dependent, the terminology "effective equation-of-state (EEOs) measurement" was used to differentiate these results from the vapor pressure of pure materials. This technique can be easily applied to plutonium bearing and previously irradiated fuels. The EEOs technique was used to determine the vapor pressure of

high purity uranium dioxide with an oxygen-to-metal ratio of 2.08. Energy depositions of up to 2720 J/g relative to room temperature were obtained, resulting in measured pressures in excess of 38 mpa.

2826 (SAND--77-1095C) LARGE SCALE SODIUM INTERACTIONS. PART 3. CHEMICAL PHENOMENA WITH LIMESTONE CONCRETE. Sallach, R.A. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 4p. (CONF-771109--27). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

The description of the chemical processes and reaction products resulting from the exposure of concrete to molten sodium metal is important for a thorough, realistic assessment of the safety of CRBR-type reactors. Concretes are in general complex heterogeneous substances whose ingredients can be derived from many sources. Consequently a wide variety of reaction processes and products might be anticipated. Initial attention has focused on a concrete in which both the aggregate and sandy components are derived from limestone. Presented are the chemical observations and experimental data from tests in which molten sodium metal at approximately 500°C is dropped into cold limestone concrete crucibles. Thermocouples immersed in the sodium pool indicate that the reaction proceeds in two stages. In the first stage which lasts 5 to 8 minutes, the temperature of the reacting mass hovers around 500°C. This stage is followed by a second stage of longer duration--greater than 100 minutes--where the temperature is 700 to 800°C. The main reaction product is a hard, fused, black slag which contains about 3/4 of the sodium in the initial charge. A secondary product is sodium oxide aerosol which accounts for the remaining 1/4 of the charge. It is significant that no free sodium metal is found in the slag; all sodium has completely reacted.

2827 (SAND--77-1096C) FIRST IN-CORE MOLTEN FUEL POOL EXPERIMENT. Plein, H.G.; Carlson, G.A. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 5p. (CONF-771109--36). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

If during a LMFBR accident a fuel debris bed is not adequately cooled by overlying sodium and dryout occurs, then the fuel would heat up and melt. Molten fuel pools are of concern due to their potential for ablation of the supporting structure and possible penetration of the reactor vessel. The major goal of the molten fuel pool program has been to develop a versatile experiment in which heat flux and structural ablation could be studied using real materials under typical temperature and heating conditions. This has now been accomplished, using the Annular Core Pulse Reactor (ACPR) to fission heat enriched UD_2 . In the first experiment conducted, a small portion of a 0.834 kg UD_2 sample was melted, and temperature data were recorded to above the melting point of the UD_2 using ultrasonic thermometry.

2828 (SAND--77-1101C) IMPORTANCE OF SCALE AND TRIGGERING EFFECTS IN VAPOR EXPLOSIONS. Williams, D.C. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 6p. (CONF-771109--16). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

An attempt is made to show one way in which

large-scale, coherent vapor explosions can be understood in terms of local interaction properties. The impossible importance the presence of a suitable triggering event may have in determining whether an explosion actually occurs on a particular occasion is noted. Certain small-scale interaction properties are assumed to exist and the large-scale system behavior these imply are explored.

- 2829 (SAND--77-1117C) LARGE SCALE TEST TECHNOLOGY. Philbin, J.S.; Pickard, P.S.; Walker, J.V.; Young, M.F.; Odom, J.P. (Sandia Labs., Albuquerque, NM (USA); Science Applications, Inc., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 4p. (CONF-771109--6). Dep. NTIS, PC A02/MF A01. From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

The paper presents an overview of conceptual designs of facilities suitable for performing large scale LMFBR reactor safety tests. The versatility offered by the approaches discussed is intended to satisfy many of the requirements for establishing a pertinent experimental data base for core disruptive accident parameters.

- 2830 (SAND--77-1134) ADVANCED REACTOR SAFETY RESEARCH PROGRAM QUARTERLY REPORT, APRIL-JUNE 1977. (Sandia Labs., Albuquerque, NM (USA)). Nov 1977. Contract EY-76-C-04-0789. 207p. (NUREG--0181-3). Dep. NTIS, PC A10/MF A01.

Information is presented concerning accident energetics; core debris behavior; sodium containment and structural integrity; research for elevated temperature design criteria; fuel motion detection; ACPR fuel motion system; and advanced reactor safety research assessment.

- 2831 (SAND--77-1157C) APPLICATION OF ULTRASONIC THERMOMETRY IN LMFBR SAFETY RESEARCH. Carlson, G.A.; Sullivan, W.H.; Plein, H.G. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 19p. (CONF-771064--2). Dep. NTIS, PC A02/MF A01.

From Ultrasonics symposium; Phoenix, AZ, USA (26 Oct 1977).

Ultrasonic thermometry has many potential applications in reactor safety research, where extremely high temperatures and lack of visual access may preclude the use of conventional diagnostics. An application (the in-core molten fuel pool experiment) will be described in which thoriated tungsten ultrasonic thermometers were used to measure temperatures in UO_2 to incipient melt (2860°). Each thermometer included five sensor elements 10 mm long, providing five temperatures within the UO_2 at various axial locations. The 10 mm spatial resolution is about five times better than previous applications of the technique. Temperature resolution of $\pm 10^\circ C$ was indicated by calibration data. Besides providing temperature data approximately 1000°C higher than were obtained with thermocouples, the thermometer yielded valuable axial temperature profile data. Details of the sensors, exciting coils, and signal conditioning electronics will be given.

- 2832 (SAND--77-1342) EULERIAN FINITE-DIFFERENCE CALCULATIONS OF EXPLOSIONS IN PARTIALLY WATER-FILLED OVERSTRONG CYLINDRICAL CONTAINMENT VESSELS. Thompson, S.L.; Herrmann, W. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 145p. Dep. NTIS, PC A05/MF A01.

Portions of this document are illegible.

Calculations, using the two-dimensional Eulerian finite-difference code CSQ, were performed for the problem of a small spherical high-explosive charge detonated in a closed heavy-walled cylindrical container partially filled with water. Data from corresponding experiments, specifically performed to validate codes used for hypothetical core disruptive accidents of liquid metal fast breeder reactors, are available in the literature. The calculations were performed specifically to test whether Eulerian methods could handle this type of problem, to determine whether water cavitation, which plays a large role in the loadings on the roof of the containment vessel, could be described adequately by an equilibrium liquid-vapor mixed phase model, and to investigate the trade-off between accuracy and cost of the calculations by using different sizes of computational meshes. Comparison of the experimental and computational data shows that the Eulerian method can handle the problem with ease, giving good predictions of wall and floor loadings. While roof loadings are qualitatively correct, peak impulse appears to be affected by numerical resolution and is underestimated somewhat.

- 2833 (SAND--77-1394) LAGRANGIAN FINITE-DIFFERENCE CALCULATIONS OF AN EXPLOSION IN A PARTIALLY WATER-FILLED RIGID CYLINDRICAL TANK. Swagle, J.W. (Sandia Labs., Albuquerque, NM (USA)). Apr 1978. Contract EY-76-C-04-0789. 77p. Dep. NTIS, PC A04/MF A01.

Calculations using the Lagrangian finite-difference code, TOODY, have been performed for the problem of an explosion in a partially water-filled rigid cylindrical tank. Experimental data for this problem formed the basis of comparison for one of a series of test problems designed to validate codes used to model reactor blowdown situations. The calculations presented here show effects of zoning variations on calculational results.

- 2834 (SAND--77-1413C) SANDIA IN-CORE MOLTEN FUEL POOL EXPERIMENT RESULTS. Plein, H.G.; Carlson, G.A. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 2p. (CONF-771108--1). Dep. NTIS, PC A02/MF A01.

From Post accident heat removal information exchange meeting; Chicago, IL, USA (2 Nov 1977).

The major initial objective of the in-core molten fuel pool program at Sandia has been to develop a versatile experiment in which relative heat fluxes and structural ablation could be studied using real materials under typical temperature and heating conditions. The initial experiment series is designed to prove containment design, test ultrasonic thermometers, and provide material interaction data. The first two molten fuel pool experiments, MP-1 and MP-2, were successfully conducted in the Annular Core Pulse Reactor (ACPR) on June 24 and July 29, 1977. These first two experiments demonstrated the adequacy of the experiment package for heating and containment of UO_2 to melt. They have shown the usefulness of ultrasonic thermometry for making multiple temperature measurements to the melting point of UO_2 , but also indicated that some problems remain to be solved in this application of the technique.

- 2835 (SAND--77-1421C) LARGE SCALE SODIUM TEST FACILITY. King, D.L.; Smaardyk, J.E.; Sutherland, H.J.; Sallach, R.A. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 8p. (CONF-771108--

4). Dep. NTIS, PC A02/MF A01.

From Post accident heat removal information exchange meeting; Chicago, IL, USA (2 Nov 1977).

The large scale sodium test facility in use at Sandia Laboratories was designed to be used for a wide range of sodium interaction experiments. Of initial concern is the interaction of hot sodium with concrete under conditions similar to accident conditions in any sodium cooled reactor. The tests to be run cover the cases of sodium spills and sprays on bare concrete and on cells with defective steel liners. The initial series of tests in progress at the facility involves rapidly dropping from 20 kg to 225 kg of sodium at temperatures from 550°C to 850°C into concrete crucibles that are representative of the concrete to be used at the Clinch River Breeder Reactor Plant (CRBRP). Specific objectives are to obtain sufficient information to completely describe the sodium/concrete interaction including: reaction rate, heat of reaction, gas evolution and flow, movement of the reaction front, reaction product formation, and the effects of variation of drop temperature and reaction area geometries.

2836 (SAND--77-1443C) IN-REACTOR EXPERIMENTS WITH SIMULATED LMFBR DEBRIS BEDS. Rivard, J.B. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 4p. (CONF-771108--2). Dep. NTIS, PC A02/MF A01.

From Post accident heat removal information exchange meeting; Chicago, IL, USA (2 Nov 1977).

A LMFBR accident in which molten core material is suddenly quenched with subcooled liquid sodium could result in extensive fragmentation and dispersal of fuel as subcritical beds of particulate debris. The primary safety concern for these debris beds is their capacity for transfer of the fission product decay heat to overlying liquid sodium by natural processes. The objective of the D-series of experiments is to closely simulate decay-heated debris and to determine the limits of stable operation wherein dryout does not occur. This is accomplished by fission heating enriched UO_2 particulate in subcooled liquid sodium in the Annular Core Pulse Reactor (ACPR). Results to date are presented.

2837 (SAND--77-1470) DESIGN CONSIDERATIONS FOR EPITHERMAL PULSE REACTORS. Ostensen, R.W. (Sandia Labs., Albuquerque, NM (USA)). Jan 1978. Contract EY-76-C-04-0789. 33p. Dep. NTIS, PC A03/MF A01.

Simplified design criteria were developed for scoping analyses of epithermal pulse reactors for use in LMFBR safety testing. By using these criteria, materials and designs were investigated to determine performance limits of moderately sized reactor cores. Several designs are suggested for further study. These are a gas-cooled core fueled with a heterogeneous mixture of Fe- UO_2 cermet and BeO- UO_2 ceramic fuels, and a heavy-water-cooled core fueled with an Fe- UO_2 cermet.

2838 (SAND--77-1483) CONCEPTUAL DESIGNS FOR LMFBR SPENT FUEL SHIPPING CASKS. Allen, G.C. Jr.; Eakes, R.G.; Freedman, J.M.; Pope, R.B.; Dupree, S.A.; Schimmel, W.P. Jr. (Sandia Labs., Albuquerque, NM (USA)). Mar 1978. Contract EY-76-C-04-0789. 44p. Dep. NTIS, PC A03/MF A01.

A spent-fuel shipping cask for use at both the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor Plant (CRBRP) is

being developed. In compliance with milestones established for the Sandia Labs LMFBR spent-fuel shipping activity, conceptual designs of an LMFBR shipping cask are described. The detailed design based on one of the concepts will commence after review and comments have been received from the cognizant organizations. Conceptual designs described in this document comply with the shipping system requirements as defined in the LMFBR Spent-Fuel Shipping System Design Description. Additional considerations which have affected the conceptual designs currently being used are: (1) the rotating basket option has been excluded; (2) spent fuel must be adequately cooled during loading and unloading in a one-atmosphere argon environment; (3) passive cooling of the cask during shipment is required; (4) six-month cooled CRBRP fuel prescribes the radiation and thermal source; and (5) the shipping cask will be transported by rail. Heat transfer requirements during loading and unloading will substantially impact the shipping system. Spent-fuel assemblies may be placed in canisters containing a non-gaseous coolant prior to shipment or may be shipped bare subject to certain loading and cooling restrictions. The recommended option is to place a liquid metal in the canister and then insert the fuel assembly.

2839 (SAND--77-1486) EVALUATION OF ORGANIC COOLANTS FOR THE TRANSPORTATION OF LMFBR SPENT FUEL RODS. Arnold, C. Jr. (Sandia Labs., Albuquerque, NM (USA)). May 1978. Contract EY-6-C-04-0789. 41p. Dep. NTIS, PC A03/MF A01.

The physical and chemical processes that are likely to occur when sodium coated LMFBR spent fuel rods are submerged in various aromatic organic coolants was defined by means of immersion experiments carried out with sodium coated 304 stainless steel coupons. Upon immersion of sodium coated coupons at 220°C in hydrocarbon type coolants such as Therminol 88, a mixture of terphenyls, not only was the metallic sodium retained on the coupon, but a carbonaceous coating formed on the surface of the sodium. In contrast, coolants that contained aromatic ether bonds, such as Dowtherm A, reacted with sodium at 220°C to form phenolate and other salts, which precipitated from the coolant in the form of a dark sludge. With Dowtherm A, removal of metallic sodium from the coupon was essentially complete in a matter of hours at temperatures of 160--220°C. Data on the rate and efficiency of sodium removal upon immersion in Dowtherm A at elevated temperatures were obtained. In addition the kinetics and chemistry of the sodium/Dowtherm A reaction were defined. Because sodium sludges are potentially incompatible with the containing structural materials and the fuel elements, it is recommended that sodium be removed prior to immersion in the coolant via reaction with benzoic acid; this method should be adaptable to the facilities at reactor sites. In aging studies Dowtherm A was found to be thermally stable up to 400°C and radiatively stable at ambient conditions. The combined effect of heat and radiation was not defined.

2840 (SAND--77-1553C) DEVELOPMENT OF A CODED APERTURE FUEL MOTION DIAGNOSTICS SYSTEM FOR THE ACPR (UPGRADE). Kelly, J.G.; Stalker, K.T. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 7p. (CONF-771210--1). Dep. NTIS, PC A02/MF A01.

From Meeting on fuel and clad motion diagnostics for fast reactor safety test facilities; Los Alamos, NM, USA (5 Dec 1977).

As part of Sandia Laboratories program to

study simulated core disruptive accidents in reactor safety research, a fuel motion detection system based on coded aperture imaging is being developed. Although fuel motion has been observed at the TREAT by the fast neutron hodoscope and with a Vidicon pinhole camera technique, the coded aperture system described offers a potential for lower cost, higher spatial resolution, three dimensional imaging, and higher frame rates at lower fluences than either of the other techniques.

- 2841 (SAND--77-1556) ELEVATED TEMPERATURE CREEP AND FATIGUE DAMAGE OF A 2.25 CR--1 MO STEEL WELDMENT. Van Den Avyle, J.A. (Sandia Labs., Albuquerque, NM (USA)). Jan 1978. Contract EY-76-C-04-0789. 35p. Dep. NTIS, PC A03/MF A01.

In weldments between dissimilar metals wide variations occur in metallurgical structure and mechanical properties, so that for good structural design it is necessary to understand the mechanical response of individual microstructural segments of the weld. This study investigates elevated temperature properties of a 2.25 Cr--1 Mo ferritic steel base metal welded with Chromenar 382V (Inconel 82) filler metal. Creep and low-cycle fatigue tests at 866°K (1100°F) show the filler metal and heat affected zone to be much stronger than the base metal. Optical microscopy does not show significant aging effects in the short-term fatigue tests or creep tests of 1180 hour duration.

- 2842 (SAND--77-1557) MICROSTRUCTURAL EVALUATION AND NON-DESTRUCTIVE EXAMINATION OF 2.25 CR--1 MO STEEL. Van Den Avyle, J.A.; Jones, W.B.; Gieske, J.H. (Sandia Labs., Albuquerque, NM (USA)). Jul 1978. Contract EY-76-C-04-0789. 61p. Dep. NTIS, PC A04/MF A01.

Specimens of 2.25 Cr-1 Mo steel were damaged to various levels by creep and combined creep-fatigue cycling at 866°K (1100°F) with hold periods. Ultrasonic shear wave velocity measurements were performed and showed small changes with large amounts of creep or fatigue damage; these changes were deemed too small for practical application. Measurements of magnetic properties were unsuccessful in following damage. Optical and transmission electron microscopy (TEM) were performed on the samples. TEM observations showed that microstructural changes which occurred at 866°K (1100°F) did not occur homogeneously throughout the matrix; during creep, carbide spheroidization and coarsening began in widely separated regions, with an increasing fraction of the material undergoing the process with time. A precipitate was observed to form during fatigue which did not occur during simple aging or creep.

- 2843 (SAND--77-1648C) FLASH X-RADIOGRAPHY FOR MATERIAL MOTION DETECTION. Choate, L.M.; Buckalew, W.H.; Posey, L.D. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 12p. (CONF-771210--2). Dep. NTIS, PC A02/MF A01.

From Meeting on fuel and clad motion diagnostics for fast reactor safety test facilities; Los Alamos, NM, USA (5 Dec 1977).

A significant part of the experimental program dealing with the behavior of prototypic LMFBR fuel pin bundles under mild to severe power transients is that of the observation of fuel/cladding/coolant motion. The feasibility of using electron beam flash x-radiography as a diagnostic tool for safety test facilities is presently under evaluation. A summary of the

objectives and approach of the flash x-radiography program is presented.

- 2844 (SAND--77-1724) APPLICATION OF CODED SOURCE X-RADIOGRAPHY TO THE OBSERVATION OF NUCLEAR FUEL MOTION. Kelly, J.G. (Sandia Labs., Albuquerque, NM (USA)). Feb 1978. Contract EY-76-C-04-0789. 59p. Dep. NTIS, PC A04/MF A01.

The observation of nuclear fuel motion in the test section of a reactor designed for fast reactor safety experiments may be possible with an intense x-ray source. A study has indicated that during peak driver core power a high power relativistic electron accelerator may be required. The coded source technique allows high spatial resolution to be recovered without serious loss in detection efficiency. The results are summarized of the first application of the method to this broad spectrum, high energy realm using the Hermes II facility (10 MeV, 100 kA, 70 ns). A reconstruction of an LMFBR nuclear fuel pin shaped object has been obtained with 3 mm spatial resolution from a source that is 1.3 cm wide and 20 cm long.

- 2845 (SAND--77-1727) REVIEW OF NUCLEAR FUEL CYCLE ALTERNATIVES INCLUDING CERTAIN FEATURES PERTAINING TO WEAPON PROLIFERATION. Williams, D.C.; Rosenstroch, B. (Sandia Labs., Albuquerque, NM (USA)). Jan 1978. Contract EY-76-C-04-0789. 51p. Dep. NTIS, PC A04/MF A01.

Largely as a result of concerns over nuclear weapon proliferation, the U.S. program to develop and commercialize the plutonium-fueled breeder reactor has been slowed down; interest in alternative fuel cycles has increased. The report offers an informal review of the various nuclear fuel cycle options including some aspects relevant to weapon proliferation, although no complete review of the latter subject is attempted. Basic principles governing breeding, reactor safety, and efficient utilization of fission energy resources (thorium and uranium) are discussed. The controversial problems of weapon proliferation and its relation to fuel reprocessing (which is essential for efficient fuel cycles) are reviewed and a number of proposed approaches to reducing proliferation risks are noted. Some representative specific reactor concepts are described, with emphasis on their development status, their potentials for resource utilization, and their implications for proliferation.

- 2846 (SAND--77-1828C) PRELIMINARY RESULTS FROM INITIAL IN-PILE DEBRIS BED EXPERIMENTS. Rivard, J.B. (Sandia Labs., Albuquerque, NM (USA)). 1977. Contract EY-76-C-04-0789. 14p. (CONF-771109--62). Dep. NTIS, PC A02/MF A01.

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

An accident in a liquid metal fast breeder reactor (LMFBR) in which molten core material is suddenly quenched with subcooled liquid sodium could result in extensive fragmentation and dispersal of fuel as subcritical beds of frozen particulate debris within the reactor vessel. Since this debris will continue to generate power due to decay of retained fission products, containment of the debris is threatened if the generated heat is not removed. Therefore, the initial safety question is the capacity which debris beds may have for transfer of the decay heat to overlying liquid sodium by natural processes--i.e., without the aid of forced circulation of the coolant. Up to the present time, all experiments on debris bed

behavior either have used substitute materials (e.g., sand and water) or have employed actual materials, but atypical heating methods. Increased confidence in the applicability of debris bed simulations is afforded if the heat is generated within the fuel component of the appropriate fast reactor materials. The initial series of in-pile tests reported on herein constitutes the first experiments in which the internal heating mode has been produced in particulate oxide fuel immersed in liquid sodium. Fission heating of the fully-enriched UO_2 in the experiment while it is contained within Sandia Laboratories Annular Core Pulse Reactor (ACPR), operating in its steady-state mode, approximates the decay heating of debris. Preliminary results are discussed.

- 2847 (SAND--77-1975) ADVANCED REACTOR SAFETY RESEARCH PROGRAM. QUARTERLY REPORT, JULY--SEPTEMBER 1977. (Sandia Labs., Albuquerque, NM (USA)). May 1978. Contract EY-76-C-04-0789. 148p. Dep. NTIS, PC A07/MF A01.

Progress is summarized in the following LMFBR-related safety studies: (1) accident energetics, (2) core debris behavior, (3) sodium containment and structural integrity, (4) research of elevated temperature design criteria, (5) fuel motion detection, (6) ACPR fuel motion system, and (7) advanced reactor safety research assessment. (DG)

- 2848 (SAND--78-0078C) MEASUREMENT OF THE PENETRATION OF LIQUID SODIUM INTO LIMESTONE CONCRETE. Sutherland, H.J.; Kent, L.A. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 7p. (CONF-780622--8). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

A pulse-echo ultrasonic system is used to monitor the movement of the interface created by pouring hot, liquid sodium onto limestone concrete. The essence of this system is to measure the time required for an acoustic signal, generated by the transmitting transducer, to traverse the specimen, reflect at the pool/concrete interface, and then retrace the sample.

- 2849 (SAND--78-0216C) NETWORK MODEL OF FREE CONVECTION WITHIN INTERNALLY HEATED POROUS MEDIA. Conrad, P.W. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 6p. (CONF-780622--6). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

Portions of document are illegible.

A hypothetical core-disruptive accident in a liquid metal fast breeder reactor may result in the formation of an internally heated debris bed which is immersed in liquid sodium. A method is demonstrated for computing the heat transfer from such a bed to the overlying sodium pool due to single-phase, free convection.

- 2850 (SAND--78-0232C) PROMPT BURST ENERGETICS EXPERIMENTS URANIUM CARBIDE SERIES. Reil, K.O.; Young, M.F.; Plitz, H. (Sandia Labs., Albuquerque, NM (USA); Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). 1978. Contract EY-76-C-04-0789. 14p. (CONF-780622--21). Dep. NTIS (US Sales Only), PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

A series of three prompt burst energetics

(PBE) experiments utilizing fresh uranium carbide fuel and stagnant sodium coolant were performed in Sandia Laboratories Annular Core Pulse Reactor (ACPR). These experiments (PBE-SG1, -SG2, and -SG3) were collaborative efforts between the Breeder Reactor Project, GfK, Karlsruhe, West Germany, and Sandia Laboratories. PBE experiments are designed to study the energetic response of various reactor fuel-clad-coolant systems under superprompt critical conditions. The direct aim is to identify and characterize the phenomena resulting in pressure generation and the conversion of thermal energy to work.

- 2851 (SAND--78-0233C) CODED APERTURE IMAGING IN MANY-PIN FUEL BUNDLES. McArthur, D.A.; Halbleib, J.A. Sr.; Morel, J.E. (Sandia Labs., Albuquerque, NM (USA)). [nd]. Contract EY-76-C-04-0789. 12p. (CONF-780622--17). Dep. NTIS, PC A01/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

As part of the experimental fast reactor safety research program at Sandia Laboratories for the Nuclear Regulatory Commission, the problems of fuel motion imaging in safety tests on a variety of facilities (ACPR UPGRADE, TREAT, TREAT UPGRADE, STF, CABRI) are being studied. Coded aperture imaging of both neutrons and gamma rays from many-pin fuel bundles is being studied. Realistic, detailed calculations are reported on gamma ray coded aperture imaging in many-pin bundles, and preliminary experiments on coded aperture imaging with fast neutrons.

- 2852 (SAND--78-0236C) SAS3A CALCULATIONS FOR FORCED CONVECTION SODIUM BOILING AT LOW POWER. Eisenhower, S.W.; Heames, T.J. (Washington Univ., Seattle (USA); Argonne National Lab., IL (USA)). 1978. Contract EY-76-C-04-0789. 7p. (CONF-780622--16). Dep. NTIS (US Sales Only), PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

Recently, sodium boiling experiments have been performed using a gamma-heated bundle in the French test reactor Rapsodie. The amount of thermal energy that can be produced with this method is small, typically 1 to 10 kW/m² pin. In terms of the thermalhydraulics involved, this type of experiment can be classified as forced convection boiling at low heat flux since, by design, a small but positive net flow exists during boiling. The characteristics (low power, small coolant mass flux, and net upward flow) are different from those normally examined. In order to better understand such tests, out-of-core experiments were performed using electrically-heated seven rod bundles at the Kernforschungszentrum Karlsruhe (KfK). The paper examines a comparison between the data obtained in the KfK experiments and calculations performed with the SAS3A computer code.

- 2853 (SAND--78-0247C) CONCEPTUAL DESIGNS OF SPENT-FUEL SHIPPING CASKS FOR THE U.S. BREEDER REACTOR TECHNOLOGY PROGRAM. Allen, G.C. Jr.; Freedman, J.M. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 18p. (CONF-780506--7). Dep. NTIS, PC A02/MF A01.

From 5. symposium on packaging and transportation of radioactive materials; Las Vegas, NV, USA (7 May 1978).

Portions of document are illegible.

Conceptual designs of spent-fuel shipping casks for use by facilities participating in the Breeder Reactor Program have been developed

by Sandia Laboratories, Albuquerque, New Mexico. Some aspects of conceptual designs that are reviewed are interface requirements and cask design features. The major emphasis of the work to date has been to develop cask designs which would be suitable for use at the Fast Flux Test Facility (FFTF) and the Clinch River Breeder Reactor Plant (CRBRP). However, the advanced technology utilized in developing spent-fuel shipping-cask designs for these facilities results in many cask features which would be required by all breeder reactor plants.

- 2854 (SAND--78-0252C) SENSITIVITY STUDY OF LMFBR CORE DISASSEMBLIES. Ostensen, R.W. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 7p. (CONF-780622--22). Dep. NTIS, PC A02/MF A01. From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

The primary concern with respect to a prompt critical excursion in an LMFBR is whether the primary containment will be damaged, leading to an early release of radioactivity. The study examines the sensitivity of the damage potential of a core disassembly to ramp rate through prompt critical; initial power level; fuel heat capacity; fuel vapor pressure.

- 2855 (SAND--78-0255C) EFFECT OF CREEP ON THERMAL FRACTURING IN OXIDE CRUSTS. Lipinski, R.J. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 7p. (CONF-780622--4). Dep. NTIS, PC A02/MF A01. From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

The stability of a fuel crust, especially on a molten sublayer such as steel, has been the subject of several papers. If the fuel crust is found to be unstable, the effect on a core-disruption accident could be far reaching: fuel would freeze more quickly in the structure above and below the core causing a plug and limiting the amount of early fuel dispersal, and high heat transfer to uninsulated walls, along with entrainment of cool steel droplets, would cool the fuel pool and reduce the amount of boilup. In a single subassembly accident, ablation of the walls would increase, thus increasing the amount of radial damage and the possibility of accident propagation. Conversely, a stable fuel crust would in general delay wall ablation and enhance upward fuel dispersal. It was recently proposed by Ostensen that a fuel crust would crack from thermal stress alone through as much as 75% of its thickness. In that paper, the effect of creep was left for future study. It is the purpose of this work to investigate that effect. A fuel crust growing on a layer of molten steel was considered.

- 2856 (SAND--78-0257C) CLOSE-IN SAMPLING OF AEROSOLS FROM FISSION VAPORIZED UO_2 FUEL. Elrick, R.M. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 3p. (CONF-780622--9). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

In order to assess the threat to public safety in the hypothesized event of a breach of containment following a core disruptive accident, it is necessary to know precisely the state and transport characteristics of the aerosols produced in the hypothesized accident. Data on the initial characteristics of these aerosols produced from neutronically heated fuel do not presently exist. The purpose of the Sandia in-pile aerosol experimental program is

to study the generation and transport of fuel debris produced from fission heating of reactor fuels as the debris interacts and is modified by structure and coolant. The paper covers the initial phase of the program in which a bare pin of fresh UO_2 was heated into the two phase liquid-vapor state by a 2380 kJ/kg 5 ms FWHM pulse. Close-in time-resolved sampling of the fuel particles established an aerosol source term for the fuel.

- 2857 (SAND--78-0266C) FIRST VISUAL IN-PILE FUEL DISRUPTION EXPERIMENTS. Cano, G.L. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 10p. (CONF-780622--11). Dep. NTIS, PC A02/MF A01. From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

A number of in-pile and out-of-pile experiments as well as analytical studies have addressed LMFBR fuel disruption: several disruption models have been discussed in the literature. Hypotheses have included, among others, that the fuel could: (1) breakup rapidly in a 'dust-cloud'; (2) swell in the solid or liquid state to a 'froth' with fission gas forming the discontinuous phase; or (3) melt and slump with release of fission gas. Sandia Fuel Disruption Experiments are described which have been designed to study this phenomenology and to clarify this issue.

- 2858 (SAND--78-0274C) FUEL PIN MODELING FOR THE PROMPT BURST EXCURSION (PBE) EXPERIMENTS. Young, M.F. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 5p. (CONF-780622--1). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

A series of in-pile experiments has been conducted at Sandia Laboratories to provide information on pressure levels and conversion of thermal energy into mechanical work in various advanced fuel-coolant systems during hypothetical, superprompt critical excursions. Identification of the phenomena causing pressure generation requires knowledge of the thermodynamic state of the fuel and coolant at the time of pin failure and fuel-coolant motion and mixing following pin failure. The paper describes the salient features of a pin model, EXPAND, used to predict pin failure time and location, and to calculate internal fuel motion and the temperature profile in the fuel-clad-coolant system up to the time of failure.

- 2859 (SAND--78-0363C) ANALYSIS OF IN-PILE FUEL DISRUPTION EXPERIMENTS. Ostensen, R.W.; Young, M.F. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 5p. (CONF-780622--3). Dep. NTIS, PC A02/MF A01.

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

For analysis of a loss-of-flow (LOF) accident in an LMFBR, knowledge of the mode of disruption of irradiated fuel is of critical importance. In addition, to determine whether the reactor can undergo a severe prompt critical power excursion it is important to know the time of onset and rate of fuel dispersal, if any. A series of experiments has been performed at Sandia Laboratories to provide the necessary data base. The failure of irradiated fuel in the ACPR reactor was recorded by high speed cinematography. The paper compares the experimental data to available fuel disruption models, and indicates the model improvements needed to reconcile the models with the data.

2860 (SAND--78-0437C) IN-REACTOR EXPERIMENTS ON THE COOLING OF FAST REACTOR DEBRIS. Rivard, J.B. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 10p. (CONF-781022--8). Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

Following a reactor accident, particulate debris beds may be formed. The primary safety question is, "Will decay-heated debris beds be cooled by the overlying sodium, or will they again melt and threaten the containment." This paper describes the first debris bed experiments using internally-heated oxide fuel in liquid sodium, summarizes the results from the first three in-reactor experiments, provides a description of an analysis of the sustained dryout phenomenon which occurred during the third experiment, and offers an important conclusion on debris bed dryout.

2861 (SAND--78-0742C) LEAKAGE EFFECTS ON LMFBR CELL LINERS. Stone, C.M.; Gartling, D.K.; Nickell, R.E. (Sandia Labs., Albuquerque, NM (USA); Pacifica Technology, Del Mar, CA (USA)). 1978. Contract EY-76-C-04-0789. 40p. (CONF-780609--4). Dep. NTIS, MF A01.

From ASME/CSME pressure vessels and piping conference; Montreal, Canada (25 Jun 1978).

Portions of document are illegible.

The thermostructural effects of a molten sodium spill onto a Liquid Metal Fast Breeder Reactor (LMFBR) cell liner are investigated utilizing a geometrically simple finite element model. The sodium spill is considered for various flow rate regimes. Because the actual characteristics of a spill are unknown, a parametric analysis was performed with the size of spill and heat transfer coefficient as variables.

2862 (SAND--78-0939C) INFLUENCE OF GAS GENERATION ON HIGH-TEMPERATURE MELT/CONCRETE INTERACTIONS. Powers, D.A. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 3p. (CONF-790111--5). Dep. NTIS, PC A02/MF A01.

From International symposium on thermodynamics of nuclear materials; Julich, F.R. Germany (29 Jan 1979).

Accidents involving fuel melting and eventual contact between the high temperature melt and structural concrete may be hypothesized for both light water thermal reactors and liquid metal cooled breeder reactors. Though these hypothesized accidents have a quite low probability of occurring, it is necessary to investigate the probable natures of the accidents if an adequate assessment of the risks associated with the use of nuclear reactors is to be made. A brief description is given of a program addressing the nature of melt/concrete interactions which has been underway for three years at Sandia Laboratories. Emphasis in this program has been toward the behavior of prototypic melts of molten core materials with concrete representative of that found in existing or proposed reactors. The goals of the experimentation have been to identify phenomena particularly pertinent to questions of reactor safety, and phenomena particularly pertinent to questions of reactor safety, and provide quantitative data suitable for the purposes of risk assessment.

CONDITIONS: I. FAST REACTOR DEBRIS BED EXPERIMENTS. II. MOLTEN CORE PENETRATION MEASUREMENTS. Rivard, J.B.; Sutherland, H.J. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 10p. (CONF-781022--6). Dep. NTIS, PC A02/MF A01.

From Meeting on nuclear power reactor safety; Brussels, Belgium (16 Oct 1978).

Following a fast reactor accident, particulate debris beds may be formed. The paper describes the first debris bed experiments using internally-heated oxide fuel in liquid sodium, summarizes the results from the first three in-reactor experiments, and offers an important conclusion on debris bed dryout. An acoustic technique is described which has been used to monitor the penetration of molten core materials into concrete. The measured initial penetration rate of the melt is of the order of 60 mm/min.

2864 (SAND--78-1249C) DEBRIS BED AND SACRIFICIAL MATERIALS INTERACTIONS AT HIGH TEMPERATURES. Plein, H.G.; Carlson, G.A. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 5p. (CONF-781105--48). Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

A series of high temperature furnace experiments were conducted to examine materials interactions under possible post-accident heat removal conditions following a hypothetical core disruptive accident. The study of material interaction within simulated dried-out debris beds and between such debris beds and candidate core retention materials produced results with significant reactor safety implications. Urania sintering dependence on particle size was examined because sintering may determine characteristics of the crust between molten fuel and structural or sacrificial material. The behavior of a mixed UO_2 /steel simulated debris bed was studied above the steel melting point. Finally the interaction of a particulate UO_2 bed with Harklase (98% MgO) brick was examined at various temperatures to determine the characteristics to be expected.

2865 (SAND--78-1251C) INTERPRETATION OF DEBRIS BED DRYOUT OBSERVED DURING THE D-3 EXPERIMENT. Rivard, J.B. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 5p. (CONF-781105--39). Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

In the D-3 experiment, a nominally 158 mm-deep particle bed was formed by 7.29 kg of UO_2 . The sodium volume fraction within the bed was 0.43. Runs were performed at bulk sodium temperatures of 773, 673, and 873 K. Dryout (liquid voidage by vapor blanketing) of a shallow zone at the bottom of the bed occurred when the power was increased in the already-boiling and convecting bed at 773 and 673 K. Temperature limitations on the experiment prevented the attainment of dryout at 873 K. Measurements and analysis of dryout observed during the D-3 experiment show: (1) the observed temperature transients during dryout can be explained in terms of a conduction process in an expanding zone; (2) the expansion is a displacement process; (3) expansion of the zone reduces the height of the overlying convection zone; and (4) while bed dryout may be a necessary condition for remelting of the fuel to occur, it may not always be a sufficient condition.

2863 (SAND--78-1223C) POST-ACCIDENT

2866 (SAND--78-1336C) TIME-RESOLVED

PENETRATION MEASUREMENTS OF LIQUID SODIUM INTO LIMESTONE CONCRETE. Smaardyk, J.E.; Sutherland, H.J.; Kent, L.A. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 9p. (CONF-781105--26). Dep. NTIS, PC A02/MF A01.

From ANS meeting; Washington, DC, USA (12 Nov 1978).

The treatment of post-accident containment problems for advanced reactors requires an understanding of how the structural concrete beneath the reactor vessel behaves when subjected to molten reactor plant materials. Of special interest is the case of a sodium spill because the resulting chemical reactions are exothermic and generate considerable quantities of gas. An important parameter in these reactions is the rate at which the reaction zone penetrates the concrete.

2867 (SAND--78-1625) CODED APERTURE IMAGING: A COMPARISON BETWEEN FRESNEL ZONE APERTURES AND UNIFORMLY REDUNDANT ARRAY APERTURES. Chu, K.W. (Sandia Labs., Albuquerque, NM (USA)). Oct 1978. Contract EX-76-C-04-0789. 61p. Dep. NTIS, PC A04/MF A01.

Using computer simulations, comparisons are made between a one-dimensional Fresnel zone aperture and a one dimensional uniformly redundant array aperture. In the former case, artifacts may produce severe distortions in the reconstructed image, and an iterative technique to remove these artifacts is described. In comparison to the Fresnel zone aperture, images reconstructed from uniformly redundant array apertures have few artifacts. A quantum noise analysis for the uniformly redundant array is presented to indicate the conditions under which a uniformly redundant array will produce a signal-to-noise advantage compared to an equivalent pinhole aperture.

2868 (SAND--78-1754C) COOLING OF PARTICULATE DEBRIS BEDS: ANALYSIS OF THE INITIAL D-SERIES EXPERIMENTS. Rivard, J.B. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 11p. (CONF-781079--1). Dep. NTIS, PC A02/MF A01.

From Postaccident heat removal information exchange; Ispra, Italy (10 Oct 1978).

In an effort to provide basic data on the cooling of fast reactor debris, three in-pile experiments employing oxide fuel particulate in liquid sodium were completed in late 1977. Preliminary results from these experiments were reported shortly after their completion at the Third Post-Accident Heat Removal Information Exchange, at Argonne National Laboratory. In these experiments, a distribution of 100 μ m to 1000 μ m-sized particles of enriched UO_2 was fission-heated to simulate decay-heated debris. In each experiment, the UO_2 particles were contained in a closed, flat-bottomed vessel 012 mm in diameter which was insulated on the diameter and bottom. Sufficient sodium was included to saturate the bed of particles and to provide a volume of bulk sodium above the bed at a controlled temperature. Parameters of interest in the experiments are given.

2869 (SAND--78-1910C) COMPARISON OF FISGAS SWELLING AND GAS RELEASE PREDICTIONS WITH EXPERIMENT. Ostensen, R.W. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 12p. (CONF-790310--4). Dep. NTIS, PC A02/MF A01.

From Specialists workshop on predictive analysis of material dynamics in LMFBR safety expts; Los Alamos, NM, USA (13 Mar 1979).

FISGAS calculations were compared to fuel swelling data from the FDI tests and to gas

release data from the FGR39 test. Late swelling and gas release predictions are satisfactory if vacancy depletion effects are added to the code. However, early swelling predictions are not satisfactory, and early gas release predictions are very poor. Explanation of these discrepancies is speculative.

2870 (SAND--78-1919C) PIN FAILURE MODELING OF THE A SERIES CABRI TESTS. Young, M.F.; Portugal, J.L. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 4p. (CONF-790310--2). Dep. NTIS, PC A02/MF A01.

From Specialists workshop on predictive analysis of material dynamics in LMFBR safety expts; Los Alamos, NM, USA (13 Mar 1979).

The EXPAND pin failure model, a research tool designed to model pin failure under prompt burst conditions, has been used to predict failure conditions for several of the A series CABRI tests as part of the United States participation in the CABRI Joint Project. The Project is an international program involving France, Germany, England, Japan, and the United States and has the goal of obtaining experimental data relating to the safety of LMFBR's. The A series, designed to simulate high ramp rate TOP conditions, initially utilizes single, fresh UO_2 pins of the PHENIX type in a flowing sodium loop. The pins are preheated at constant power in the CABRI reactor to establish steady state conditions (480 w/cm at the axial peak) and then subjected to a power pulse of 14 ms to 24 ms duration (PWHM).

2871 (SAND--78-2085C) FUEL PIN MODELING OF THE FRESH OXIDE/SODIUM PROMPT BURST ENERGETICS EXPERIMENTS. Young, M.F. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 7p. (CONF-790310--1). Dep. NTIS, PC A02/MF A01.

From Specialists workshop on predictive analysis of material dynamics in LMFBR safety expts; Los Alamos, NM, USA (13 Mar 1979).

A series of in-pile experiments has been conducted at Sandia Laboratories to provide information on pressure levels and conversion of thermal energy into mechanical work in various advanced fuel-coolant systems during hypothetical, superprompt critical excursions. Identification of the phenomena causing pressure generation requires knowledge of the thermodynamic state of the fuel and coolant at the time of pin failure and fuel-coolant motion and mixing following pin failure. Experimental data from the UO_2/Na Prompt Burst Energetics (PBE) series are compared to the results obtained from a pin model, EXPAND, used to predict pin failure time and location, internal fuel motion and the temperature profile in the fuel-clad-coolant system up to the time of failure. The model uses a Larson--Miller Parameter-life fraction rule criterion to determine failure.

2872 (SAND--78-2140C) MEASUREMENT AND ANALYSIS OF TRANSIENT VAPORIZATION IN OXIDE FUEL MATERIALS. Gorham-Bergeron, E.; Benson, D.A. (Sandia Labs., Albuquerque, NM (USA)). 1978. Contract EY-76-C-04-0789. 25p. (CONF-790111--4; IAEA-SM--236/60). Dep. NTIS, PC A02/MF A01.

From International symposium on thermodynamics of nuclear materials; Julich, F.R. Germany (29 Jan 1979).

A series of experiments is described in which samples are heated to produce high vapor pressure states in times of 10^{-6} to 10^{-3} seconds. Experimental measurements of vapor

pressures over fresh UO_2 from the pulsed electron beam and pulsed reactor heating tests are presented and compared with other high temperature data. The interpretation of the vapor pressures measured in the tests is discussed in detail. Effects of original sample stoichiometry, chemical interactions with the container and non-equilibrium evaporation due to induced temperature gradients are discussed. Special attention is given to dynamic behavior in rapid heating and vaporization of the oxide due to chemical nonequilibrium. Finally, similar projected reactor experiments on irradiated fuel are described and vapor pressure predictions made using available equilibrium models. A discussion of information accessible from such future tests and its importance is presented.

- 2873 (SAND--78-2191) DEFORMATION OF THE SANDIA METALLIC RING SEAL TEST FIXTURE. Sagartz, M.J. (Sandia Labs., Albuquerque, NM (USA)). Jan 1979. Contract EY-76-C-04-0789. 19p. Dep. NTIS, PC A02/MF A01.

A fixture being used at Sandia Laboratories Albuquerque for testing a metallic ring-type seal is described and analyzed. Formulas for the deformation of the fixture are derived based on a simple mathematical model. Numerical results are presented for a set of parameters typical of an actual test. The seal is applicable to LMFBR spent fuel shipping casks.

- 2874 (SAND--79-0057C) STUDY OF THE DISPERSIVE POTENTIAL OF IRRADIATED FUEL USING IN-CORE EXPERIMENTS. Worledge, D.H.; Cano, G.L. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 10p. (CONF-790816--42). Dep. NTIS, PC A02/MF A01. From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Under prompt burst conditions the work potential of expanding fuel vapor is a sensitive function of the ability of fission product gases to cause fuel dispersal milliseconds before dispersal would occur from fuel vapor pressure alone. This relation is of potentially significant importance and merits investigation. A fuel disruption experiment at Sandia Laboratories exhibited rapid disruption probably below but near the fuel melt temperature. The fuel was being heated at 130K/msec when vigorous dispersal occurred in the hottest regions near the fuel melting point. Fuel vapor could not have been responsible. Dispersal is attributed to rapid release of fission product gases under the prompt burst conditions. Further tests are planned to explore this important phenomenon.

- 2875 (SAND--79-0079C) PROMPT BURST ENERGETICS IN THE OXIDE/SODIUM SYSTEM. Reil, K.O.; Young, M.F. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 10p. (CONF-790816--34). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

A series of twelve Prompt Burst Energetics (PBE) experiments utilizing fresh uranium dioxide fuel pins in stagnant sodium coolant has been performed in Sandia Laboratories' Annular Core Pulse Reactor (ACPR). Results and analysis described in the paper include: observation of FCIs (pressures up to 32 MPa) in the UO_2/Na system, some apparently triggered by small pressure transients (2 MPa); prediction of failure times via the pin model EXPAND; observed thermal-to-mechanical energy

conversion ratios up to approximately 0.4%; and identification of potential reactivity effects caused by the pre- and post-failure motion of fuel.

- 2876 (SAND--79-0081C) SUMMARY OF THE FIRST THREE IN-CORE PAHR MOLTEN FUEL POOL EXPERIMENTS. Plein, H.G.; Lipinski, R.J.; Carlson, G.A.; Varela, D.W. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 11p. (CONF-790816--52(Summ.)). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Results from the first three Sandia in-core molten pool experiments are reviewed. Incipient melt of a fully enriched dry bed of UO_2 was achieved in the first experiment while 30g of UO_2 was melted in the second. The melting region developed a void surrounded by a dense crust as indicated by post-test x-radiographs. In the third experiment, a steel layer was melted by an overlying layer of UO_2 . Post-test x-radiographs showed no obvious penetration of the steel into the UO_2 porous matrix. Ultrasonic thermometry was used successfully to measure temperatures up to 3100°K. The thermal histories of the in-core experiments were satisfactorily modeled using conduction theory only. Disassembly of the experiment packages will be performed shortly.

- 2877 (SAND--79-0082C) STUDY OF HEAT TRANSFER FROM A FLOWING LIQUID TO A MELTING WALL. Eisenhower, S.W.; Lee, D.O.; Corradini, M.L.; Ostensen, R.W.; Gonzalez, F. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 23p. (CONF-790816--36). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

In order to investigate fuel freezing phenomenology in the transition phase of an LMFBR accident, a separate effects experiment was conducted. These tests investigated the melting behavior of an ice pipe due to warm refrigerant (R-11) flow. For turbulent flow, the water film is not entrained and the Nusselt number is not sensitive to the Reynolds number for a fixed mass flow rate. A thin film heat transfer model shows good agreement with the experimental results. When this thin film model is applied to the transient freezing and melting process in a fuel-steel LMFBR system, the predicted fuel penetration distance is in very good agreement with the experimental results of Spencer and Henry. This is an improvement over past fuel freezing models.

- 2878 (SAND--79-0091C) DEBRIS BED HEAT REMOVAL MODELS: BOILING AND DRYOUT WITH TOP AND BOTTOM COOLING. Lipinski, R.J.; Rivard, J.B. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 14p. (CONF-790816--40). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

A one-dimensional deep bed model for incipient dryout is developed which predicts deep bed behavior but not shallow bed behavior. A fluidization criterion results which indicates that a debris bed cannot be fluidized by the vapor flow. However, the maintenance of channels within the bed is predicted. A top-subcooled bed model is presented and evidence (both experimental and calculational) is given that the top-subcooled zone removes heat in series (not in parallel) with the boiling zone

- for sodium-cooled systems. The Sandia D-2 and D-3 top-subcooled dryout data is expressed in terms of non-subcooled dryout and compared with experimental data and model predictions from the literature. The data agrees with several predictions depending on whether the dryout in the boiling zone is dependent on the overlying subcooled zone thickness. A bottom-subcooled bed model is also presented. A post-dryout model is developed to explain the stable dry zone observed in the D-3 experiment and agreement with the data is found. Finally, the top-subcooled and post-dryout models developed are used to make predictions for the upcoming D-4 experiment.
- 2879 (SAND--79-0107C) ANALYSIS OF SANDIA IN-PILE EOS EXPERIMENTS. Breitung, W.; Gorham-Bergeron, E.; Murata, K.K. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 10p. (CONF-790816--45). Dep. NTIS, PC A02/MF A01.
From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).
Preliminary analysis has been carried out of the dynamic in-pile equation-of-state measurements for UO_2 , conducted at Sandia Laboratories, aimed at reducing the uncertainties in the effective UO_2 enthalpy corresponding to the measured pressures. Of the remaining width of the p-H band of some 350 J/g, about 200 J/g originate in the uncertainties of the analytical modelling and about 150 J/g result from the scatter in the experimental data.
- 2880 (SAND--79-0112C) PROMPT BURST ENERGETICS (PBE) EXPERIMENT ANALYSES USING THE SIMMER-II COMPUTER CODE. Tomkins, J.L.; Hitchcock, J.T.; Young, M.F. (Los Alamos Scientific Lab., NM (USA); Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 10p. (CONF-790816--47). Dep. NTIS, PC A02/MF A01.
From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).
Two of the Prompt Burst Energetics (PBE) in-pile experiments conducted at Sandia Laboratories PBE-5S and PBE-SG2, have been investigated with SIMMER-II. These two tests utilize fresh uranium oxide and fresh uranium carbide pins, respectively, in stagnant sodium. The purpose of the analysis is to investigate the applicability of SIMMER-II to this type of experiment. Qualitative agreement with measured data is seen for PBE-5S. PBE-SG2 results agree somewhat less well but demonstrate SIMMER-II's potential for describing fuel-coolant-interactions with further model development.
- 2881 (SAND--79-0123C) FUEL-COOLANT INTERACTION-PHENOMENA UNDER PROMPT BURST CONDITIONS. Jacobs, H.; Young, M.F.; Reil, K.O. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.); Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 9p. (CONF-790816--33). Dep. NTIS, PC A02/MF A01.
From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).
The Prompt Burst Energetics (PBE) experiments conducted at Sandia Laboratories are a series of in-pile tests with fresh uranium oxide or uranium carbide fuel pins in stagnant sodium. Fuel-coolant-interactions in PBE-9S (oxide/sodium system) and PBE-SG2 (carbide/sodium) have been analyzed with the MURTI parametric FCI code. The purpose is to gain insight into possible FCI scenarios in the experiments and sensitivity of results to input parameters. Results are in approximate agreement for the second (triggered) event in PBE-9S (32 MPa peak) and the initial interaction in PBE-SG2 (190 MPa peak).
- 2882 (SAND--79-0126C) MEDIUM-SCALE MELT-SODIUM FRAGMENTATION EXPERIMENTS. Chu, T.Y.; Beattie, A.G.; Drotning, W.D.; Powers, D.A. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 15p. (CONF-790816--35). Dep. NTIS, PC A02/MF A01.
From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).
The results of a series of fragmentation experiments involving up to 20 Kg of thermitically produced high temperature melts and 23 Kg of sodium are presented. Except for one experiment where some centimeter size particles are observed, the fragment distributions seem to be in the range of previous data. Spatial distribution of the fragments in the debris bed appears to be stratified. Scanning electron micrographs of fragments indicate fragmentation to be occurring in the molten state for the more intense interactions observed. Interaction data obtained show quiescent periods of 0.5 to 1.5 second between pressure pulses. The force impulse values per unit mass of melt seems to be in the same range as previous experiments.
- 2883 (SAND--79-0621) ULTRASONIC THERMOMETRY SYSTEM FOR MEASURING VERY HIGH TEMPERATURES IN REACTOR SAFETY EXPERIMENTS. Carlson, G.A.; Sullivan, W.H.; Plein, H.G.; Kerley, T.M. (Sandia Labs., Albuquerque, NM (USA)). Jun 1979. Contract EY-76-C-04-0789. 83p. Dep. NTIS, PC A05/MF A01.
Ultrasonic thermometry has many potential applications in reactor safety experiments, where extremely high temperatures and lack of visual access may preclude the use of conventional diagnostics. This report details ultrasonic thermometry requirements for one such experiment, the molten fuel pool experiment. Sensors, transducers, and signal processing electronics are described in detail. Axial heat transfer in the sensors is modelled and found acceptably small. Measurement errors, calculations of their effect, and ways to minimize them are given. A rotating sensor concept is discussed which holds promise of alleviating sticking problems at high temperature. Applications of ultrasonic thermometry to three in-core experiments are described. In them, five 10-mm-length sensor elements were used to measure axial temperatures in a UO_2 or UD_2 -steel system fission-heated to about 2860°C.
- 2884 (SAND--79-0740C) PROMPT BURST ENERGETICS (PBE) STUDIES. Reil, K.O.; Young, M.F. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 16p. (CONF-790429--2). Dep. NTIS, PC A02/MF A01.
From 4. specialist meeting on fuel-coolant interaction in nuclear reactor safety; Bournemouth, UK (2 Apr 1979).
These studies involve an in-pile experimental program and complementary analytical investigation of the energetic response of important reactor fuel-clad-coolant systems subjected to energy deposition conditions associated with super-prompt critical excursions. The objectives of the present program include identification and characterization of the phenomena which dominate the conversion of thermal energy to

work, development of models which accurately predict the energetics associated with such hypothetical accidents, and provision of the necessary input data for those models. This program includes the examination of the integral effects of fuel-clad-coolant interactions, fission gas release, and fuel and fission product vapor pressure during super-prompt critical core disruptive conditions, and thus, serves to define the initial conditions for hydrodynamic expansion of the disrupted core. Progress to date is summarized.

- 2885 (SAND--79-0752C) FCI POTENTIAL OF OXIDE AND CARBIDE FUELS: RESULTS OF THE PROMPT BURST SERIES AT SANDIA LABORATORIES. Young, M.F.; Murata, K.K.; Hitchcock, J.T.; Camp, W.J. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 13p. (CONF-790429-4). Dep. NTIS, PC A02/MF A01.

From 4. specialist meeting on fuel-coolant interaction in nuclear reactor safety; Bournemouth, UK (2 Apr 1979).

As part of an effort to establish the work potential of LMFBR fuel-coolant pairs under accident conditions, a series of prompt burst energetics (PBE) experiments has been carried out at Sandia Laboratories. In the PBE series FCI's have been observed both between UO_2 and Na and between UC and Na. As anticipated, FCI energetics are found to be greater for UC/Na. Preliminary conclusions from analyses of that series are reviewed herein. In addition to PBE analysis, work on modeling fuel fragmentation is briefly described. Finally, additional FCI-related experiments at Sandia are discussed.

- 2886 (SAND--79-1366) THERMAL EVALUATION FACILITY FOR LMFBR SPENT FUEL TRANSPORT. Wesley, D.A. (Sandia Labs., Albuquerque, NM (USA)). Apr 1980. Contract AC04-76DP00789. 107p. NTIS, PC A06/MF A01.

A full-scale mock-up of a 217 pin breeder reactor fuel assembly in a cylindrical pipe was initially designed and constructed by Oak Ridge National Laboratory (ORNL). It was transferred to Sandia where it was extensively redesigned and modified. The 217 pin hexagonal core assembly was installed in a smaller diameter stainless steel pipe which more closely represents the diameter of a shipping canister or shipping cask basket wall. Two-hundred four of the tubes are electrically heated over an active length of 4-feet and the remaining thirteen are instrumented with multiple junction thermocouples which can be traversed axially. Thermocouples and heat-flux gauges are located on the hex core and canister perimeters at several axial locations.

- 2887 (SAND--79-1602C) LMFBR ACCIDENT DELINEATION STUDY: APPROACH AND PRELIMINARY RESULTS. Williams, D.C.; Sholtis, J.A.; Rios, M.; Worledge, D.H.; Conrad, P.W.; Varela, D.W.; Pickard, P.S. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 11p. (CONF-790816--27). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Event trees have been constructed for all phases of LMFBR accidents. The trees proved useful for identifying meaningful initiating accident categories and containment responses. In these areas, quantification appears feasible, given an adequate data base. Event trees were also used to represent in-core phenomenological questions governing accident progression and energetics, but here quantification appears impracticable because

pervasive phenomenological uncertainties exist. Infrequent accident initiation is the dominant factor in assuring low risk. Nevertheless, containment promises an additional measure of risk reduction provided severe energetics are highly unlikely. The delineation served to systematize LMFBR safety issues and should aid in evaluating LMFBR R and D priorities.

- 2888 (SAND--79-1685C) SURVEY OF MELT INTERACTIONS WITH CORE RETENTION MATERIAL. Powers, D.A. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 11p. (CONF-790816--26). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

A survey of the interactions of up to 220 kg stainless steel melts at 1973°K with the candidate core retention materials borax, firebrick, high alumina cement, and magnesia is described. Data collected for the interactions include rates of material erosion, aerosol generation, gas evolution, and upward heat flux. Borax acts as an ablative solid that rapidly quenches the melt. Firebrick is ablated by the steel melt at a rate of 8.2×10^{-6} m/s. High alumina cement is found to be an attractive melt retention material especially if it can be used in the unhydrated form. Magnesia is also found to be an attractive material though it can be eroded by the molten oxides of steel.

- 2889 (SAND--79-1693) HTFLO: A COMPUTER MODEL OF A RESISTIVELY-HEATED UO_2 PIN WITH A RATE-DEPENDENT HEAT CAPACITY. Smaardyk, J.E. (Sandia Labs., Albuquerque, NM (USA)). Feb 1980. Contract EY-76-C-04-0789. 40p. Dep. NTIS, PC A03/MF A01.

A one-dimensional model is presented to describe the thermal behavior of an uranium dioxide (UO_2) rod that is heated resistively and cooled by surface radiation. A standard forward differencing scheme is used for the heat transfer calculations. Temperature-dependent electrical and thermal properties are calculated at the beginning of each time step. In addition, the heat capacity is calculated from a dynamic enthalpy model based on equilibrium and instantaneous heat capacity models with a user-selectable relaxation time. This report describes the finite difference equations used, the treatment of the source and boundary terms, and the dynamic enthalpy model. Detailed input instructions are provided and a sample calculation is shown.

- 2890 (SAND--79-1708C) KINETICS OF FISSION-VAPORIZED UO_2 FUEL DETERMINED FROM SAMPLED FUEL DEBRIS. Elrick, R.M. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 9p. (CONF-790816--46). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

Close-in time-resolved sampling of fuel aerosols, high speed photography, and photometric measurement of fuel temperatures were used to determine the thermal and kinetic characteristics of UO_2 fission heated to vaporization with a single 5 msec FWHM pulse of 2380 KJ/kg. From these data it was possible to obtain energy temperature relations and debris kinetics including particle size, velocity, density, momentum and pressure. These results demonstrate techniques for study of dynamic interactions involving liquid and vapor between fuel, cladding, coolant and structure and

produce data important for determining initial HCDA aerosol formation.

- 2891 (SAND--79-1733C) MODEL AND COMPUTER CODE FOR ENERGY AND MASS TRANSPORT IN DECOMPOSING CONCRETE AND RELATED MATERIALS. Knight, R.L. & Beck, J.V. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 6p. (CONF-790816--54). Dep. NTIS, PC A02/MF A01.

From International meeting on fast reactor safety technology; Seattle, WA, USA (19 Aug 1979).

In an HCDA, it is postulated that liquid metal coolants and more materials come in contact with the retaining concrete structure. A mathematical model and an associated computer program have been developed to describe the transient heat and mass transfer in the concrete. The program is unique in that it includes: (1) chemical reactions involving free water, chemically bound water and CO_2 ; (2) an implicit calculational procedure; and (3) change of the porosity as the concrete is decomposed. Predictions of the model are compared with experimental results obtained at Sandia Laboratories and at Hanford Engineering Development Laboratory.

- 2892 (SAND--79-2319A) HIGH-TEMPERATURE ELECTRONICS: AN OVERVIEW. Heckman, R.C. (Sandia Labs., Albuquerque, NM (USA)). 1979. Contract EY-76-C-04-0789. 8p. Dep. NTIS, PC A02/MF A01.

A summary is presented providing an overview of contemporary high-temperature electronics and identifying the major areas where developments are needed and the laboratories where research is being conducted. The geothermal program, high-temperature oil and gas well logging, jet engine monitors, and circuits for operation in the sodium coolant loop of the Clinch River Breeder reactor have stimulated research. (FS)

- 2893 (SAND--80-0026) CODED APERTURE IMAGING SYSTEM FOR NUCLEAR FUEL MOTION DETECTION. Stalker, K.T.; Kelly, J.G. (Sandia Labs., Albuquerque, NM (USA)). 1980. Contract EY-76-C-04-0789. 11p. (CONF-800453--1). Dep. NTIS, PC A02/MF A01.

From International optical computing conference; Washington, DC, USA (7 Apr 1980).

A Coded Aperture Imaging System (CAIS) has been developed at Sandia National Laboratories to image the motion of nuclear fuel rods undergoing tests simulating accident conditions within a liquid metal fast breeder reactor. The tests require that the motion of the test fuel be monitored while it is immersed in a liquid sodium coolant precluding the use of normal optical means of imaging. However, using the fission gamma rays emitted by the fuel itself and coded aperture techniques, images with 1.5 mm radial and 5 mm axial resolution have been attained. Using an electro-optical detection system coupled to a high speed motion picture camera a time resolution of one millisecond can be achieved. This paper will discuss the application of coded aperture imaging to the problem, including the design of the one-dimensional Fresnel zone plate apertures used and the special problems arising from the reactor environment and use of high energy gamma ray photons to form the coded image. Also to be discussed will be the reconstruction techniques employed and the effect of various noise sources on system performance. Finally, some experimental results obtained using the system will be presented.

- 2894 (SAND--80-0217C) DEBRIS BED EXPERIMENT NO. 4: A PRELIMINARY LOOK. Gronager, J. (Sandia Labs., Albuquerque, NM (USA)). 1980. Contract EY-76-C-04-0789. 14p. (CONF-800607--8). Dep. NTIS, PC A02/MF A01.

From American Nuclear Society annual meeting; Las Vegas, NV, USA (8 Jun 1980).

The experiment described is the fourth in a series of experiments designed to provide quantitative information on the behavior of debris beds formed from reactor materials following a core-melt accident. The D-series of experiments utilizes liquid sodium, particulate UO_2 (and in later experiments, stainless steel) to form debris beds which are fission heated, within the ACRR, to simulate fission-product heating of debris beds. The data obtained from the experiments will provide the basis of debris bed modeling in order to ascertain containment requirements.

- 2895 (SERI/TP--31-248(Vol.2)(Pt.2), pp 633-639) HIGH TEMPERATURE CORROSION FATIGUE. Indig, M.E. (General Electric Co., Pleasanton, CA). Oct 1979.

From Reliability of materials for solar energy workshop; Denver, CO, USA (18 Dec 1978).

CORROSION FATIGUE; STEAM; HEAT EXCHANGERS; LMFBR TYPE REACTORS; NUCLEATE BOILING; STEAM GENERATORS; CRACKS; INCOLOY 800; TEST FACILITIES; CHROMIUM-MOLYBDENUM STEELS; SOLAR THERMAL POWER PLANTS

- 2896 (SRD-R--82) REVIEW OF AEROSOL PROBLEMS AND THE THEORY OF AEROSOL PHYSICS WITH PARTICULAR REFERENCE TO SODIUM COOLED FAST REACTORS. Williams, R.J. (UKAEA Risley Nuclear Power Development Establishment, Culcheth. Safety and Reliability Directorate). Jan 1978. 35p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Processes that would govern the development, transport, and removal of aerosols, which are of interest in the study of hypothetical core disruptive situations in pool type sodium cooled fast reactors, are discussed. Theoretical descriptions of these processes are presented and known inadequacies indicated. The interpretation of experimental data and numeric solution of the governing equations is briefly considered.

- 2897 (SRD-R--159) EFFECT OF UNCERTAINTIES IN THERMOPHYSICAL PROPERTIES OF UO_2 AND SODIUM ON MFCI'S. MacInnes, D.A.; Martin, D.; Vaughan, G.J. (UKAEA Risley Nuclear Power Development Establishment, Culcheth. Safety and Reliability Directorate). Jul 1979. 13p. Dep. NTIS (US Sales Only), PC A09/MF A01.

Work performed under United States-UKAEA Fast Reactor Exchange Program.

Recent work has led to the values of the physical properties of molten UO_2 and sodium to be questioned. The effect of these uncertainties on MFCI's is investigated - in particular, how the interface temperature/spontaneous nucleation temperature criterion for an interaction to occur and its yield are changed. It is concluded that it is not possible to exclude MFCI's between UO_2 and sodium by this criterion given the present uncertainties.

- 2898 (STI/PUB--490) DESIGN, CONSTRUCTION, AND OPERATING EXPERIENCE OF DEMONSTRATION LMFBRs. (International Atomic Energy Agency, Vienna (Austria)). Dec 1978. 882p. (In several languages). (CONF-780407--). IAEA, Vienna, Austria, \$74.00.

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

Separate abstracts are included for each of the papers presented concerning the design and operation of LMFBR type reactors. Seven papers have previously been included in the data base.

2899 (STUDSVIK-K2--79/90) NEUTRONICS CALCULATIONS WITH EQUICYCLE VERSION 3 ON TWO HETEROGENEOUS LMFBR CONCEPTS. Haeggblom, H. (Aktiebolaget Atomenergi, Studsvik (Sweden)). Jan 1979. 14p. Dep. NTIS (US Sales Only), PC A02/MF A01.

The single-annulus LMFBR design introduced by the UKAEA Reactor Group, Risley, and the multicore concept "Bulls-eye" for which data were obtained from ORNL, have been studied using the diffusion program EQUICYCLE, developed in Studsvik. Two cross section libraries were used, namely, an adjusted set of data processed from ENDF/B-III and a set with the most important nuclides processed from ENDF/B-IV and the other data from ENDF/B-III. For the singleannulus design, an equilibrium burn-up composition was given. For the "Bulls-eye" concept, both a fresh core and an equilibrium cycle were studied. The $k_{sub}(eff)$ values obtained seem to agree approximately with other calculations using ENDF/B-IV. The adjusted cross section set gives lower breeding ratios than other data. The void coefficients were not very sensitive to the data sets used.

2900 (TI--00-LME-014) AI/LMEC EXPERIENCE WITH FREEZE SEAL AND BELLows SEAL VALVES. Freede, W.J.; Thorel, J.P. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 1977. Contract EY-76-C-03-0700. 20p. AT.

A statistical and process review was conducted regarding operational success and failure experience with freeze seal and bellows seal valves at the four principal former and present AI/LMEC sodium facilities (LCTL, SCTI, SRE, and HNPF). At three of these facilities, bellows seal valves had predominated in the initial construction. However, numerous bellows failures, sometimes accompanied by significant fire damage, soon led to large-scale conversions to freeze seal valves. One conclusion reached was that bellows seal valves simply couldn't be used for throttling purposes, as vibration and chatter on the valve disks would cause fatigue failure in the bellows.

2901 (TI--00-LME-023) SUMMARY OF LMEC SODIUM VALVE OPERATING HISTORY. Snair, G.L.; Mirless, G.; Dewart, W. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 14 Nov 1977. Contract EY-76-C-03-0700. 43p. AT.

LMEC/AI experience with valves for sodium service extend 27 years with many hundreds of thousands of valve service hours. A history of the sodium valves at the LMEC site from 1959 through 1971 was reported by Messrs Freede and Thorel. A continuation of this history to the present is presented herein. History of the more than 450 sodium valves in use during this period from 1971 to the present has included refurbishing more than 50 valves. Of these 50, less than half have been failures, and the rest have been routine overhaul or design modification. During this span since 1971 fifteen new procurement contracts have been placed for a total of 230 sodium service valves. Four special sodium valve tests involving seven test valves have also been run since 1971.

2902 (TI--04-LME-123) CALIBRATION OF 10" PM FLOWMETER IN SCTI: FRC-100E. DeVita, V. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 9 Sep 1976. Contract EY-76-C-03-0700. 100p. AT.

Portions of document are illegible.

The calibration is presented of a 10 inch PM flowmeter designated as FRC-100E at SCTI. The development of a flow factor that could be applied to the PM flowmeter measurements is described.

2903 (TID--3333-S6) LIQUID METAL FAST BREEDER REACTORS. A BIBLIOGRAPHY, JANUARY--DECEMBER 1977. Galde, D.O. (comp.). (Department of Energy, Oak Ridge, TN (USA). Technical Information Center). May 1978. 518p. Dep. NTIS, PC A22/MF A01.

Included are 2334 abstracts of LMFBR references compiled from the DOE Energy Information Data Base (EDB) covering the period January 1977 through December 1977. Corporate, Personal Author, Subject, and Report Number Indexes are provided. (DG)

2904 (TID--27701/5) LIQUID METAL FAST BREEDER REACTOR CONCEPTUAL PLANT DESIGN 1000 MWE. VOLUME V. COST/SCHEDULE. McDonald, J.S.; Andonedis, A.; Matte, J. III. (Atomics International Div., Canoga Park, CA (USA); Burns and Roe, Oradell, NJ (USA). Breeder Reactor Div.). May 1977. Contract EY-76-C-03-1142; EY-76-C-3-1145. 100p. (FBR--77-3(Vol.5)). AT.

An evaluation of the cost and schedule requirements for development of the 1000 MW(e) PLBR plant design is presented.

2905 (TID--27798) DEVELOPMENT OF LICENSING REQUIREMENTS FOR LMFBRs: A REACTOR DESIGNER'S VIEW. Bradbury, P.; Huang, H.C. (Westinghouse Electric Corp., Madison, PA (USA)). 1977. Contract EY-76-C-15-2395. 10p. Dep. NTIS, PC A02/MF A01.

The framework of rules and regulations for nuclear power plant licensing in the U.S. has been developed and established mainly through the experience in LWR licensing. Because of the important differences in technology and design between LMFBR and LWR plants, development of specific new requirements and criteria and appropriate modification of old ones must be undertaken such that the established licensing regulations can be equally and fairly implemented for commercial LMFBR licensing. The paper presents a broad overview of U.S. licensing rules and regulations and provides a review of licensing criteria, standards and ongoing development effort, with a focus on commercial LMFBR licensing needs. The review results are presented and discussed. In addition, the selection of design basis accidents and the treatment of events beyond Design Base are also reviewed and discussed.

2906 (TID--27816) DISC-DONUT-TUBE WEAR TEST REPORT, PHASE I. Kowal, K.; Knaus, S.E. (Atomics International Div., Canoga Park, CA (USA)). Jun 1976. Contract EY-76-C-03-0624-014. 77p. Dep. NTIS, PC A05/MF A01.

The report describes a test program which simulated the wear-inducing conditions in the AI Prototype CRBR Steam Generator. This was accomplished by simulating the wear inducing loading and motion of a steam tube against "disc-donut" tube spacer plates. It was found that 2-1/2 Cr-1 Mo tubes, wearing against 2-1/2 Cr-1 Mo tube spacer plates, seized and galled

as deep as .017 inches. Inconel 718 tube spacer plates uniformly wore the tubes as deep as .012 in. Aluminum bronze inserts wore as deep as .003 inches into the tube.

- 2907 (TID--27817) LLTR SERIES I TEST REQUEST. REVISION 4. GE SPECIFICATION NO. 22A3924, REV. 4. Regimbal, J.J. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). Jun 1977. Contract EY-76-C-03-0893-010. 160p. AT.

Series I tests in the Large Leak Test Rig (LLTR) using the AI-MSG (modified for large-leak testing) are requested for the LLTR test facility currently being constructed at LMEC by Atomics International. The requested program of testing provides for water and/or steam to be injected into sodium under near prototype conditions. Variables in the sodium/water reaction tests include leak size, steam generator type (evaporator or superheater) conditions, leak location, and sodium temperature. Test conditions and data acquisition requirements and methods are described in sufficient detail to serve as a basis for preparing the full scope of testing procedures necessary to accomplish the program objectives.

- 2908 (TID--27821) VIBRATION TESTS OF A FULL SCALE WATER MODEL OF A CLINCH RIVER STEAM GENERATOR MODULE. Durand, R.E.; Gabler, M.J.; Weinberg, L.; Yang, T.M. (Atomics International Div., Canoga Park, CA (USA); General Electric Co., Sunnyvale, CA (USA)). [nd]. Contract EY-76-C-15-2395. 17p. Dep. NTIS, PC A02/MF A01.

Cross and parallel flow induced vibrations were measured in a shortened full scale model of the CRBRP steam generators using water as the working fluid. Measurements were made on five selected tubes, some of which fit through oversize tube support holes. Power spectral density plots were made of vibration parameters at selected locations. Maximum displacement was about 3 mils and occurred at the lower frequencies. The maximum vibrational stress was calculated to be 2000 psi. Measurements of vortex shedding frequencies showed that vortices are not a dominant source of tube excitation. In general, no adverse effects on the CRBRP steam generators were identified which affect service life or structural integrity.

- 2909 (TID--27889) SEISMIC DESIGN STUDY ON CANTILEVER CYLINDRICAL SHELLS. Wu, J.Na.C.; Cory, J.F. Jr. (Babcock and Wilcox Co., Akron, OH (USA). Nuclear Equipment Div.). 1975. Contract EY-76-C-15-0003. 21p. Dep. NTIS, PC A02/MF A01.

One way to design a nuclear steam generating system to withstand seismic stresses is to establish a mathematical model, to which seismic loading can be applied, to determine the dynamic responses of the system. A practical approach for establishing such a model is to idealize the components in terms of beam elements which form the essential building blocks of the analytical model. The major shell structures of reactor vessels and steam generators are usually supported at a single elevation in the system, and are often simulated by cantilever beam elements. The use of a beam element to represent shell structure must be justified. The paper presents the results of a parametric study of thin-walled cylindrical shells which are clamped at one end while the other end is free. By using 3-D plate/shell finite element analysis, the region of application, where the beam element can

reasonably simulate the shell structure, has been defined as the combination of its geometrical parameters, e.g., the thickness/radius and length/radius. As an illustrative example, the 3-D finite element method has been used to investigate the seismic analysis of a cantilever cylindrical shell similar to the Clinch River LMFBR reactor vessel. The numerical values are compared with those produced by a beam element model. The results and conclusions obtained by this investigation can be applied to improve dynamic modeling techniques in general.

- 2910 (TID--27895) STANDPIPE-BUBBLER PUMP LEVEL CONTROL STUDY SYSTEM DESIGN LAYOUT (TASK A) W.P. 4413. ENGINEERING MEMORANDUM NO. 0153. Salant, R.F.; Cook, M.E. (Borg-Warner Corp., Los Angeles, CA (USA). Byron Jackson Pump Div.). 29 Apr 1975. Contract EY-76-C-03-0893-001. 125p. AT.

Pump tank liquid sodium level control within the specified normal and transient system operating modes utilizing a Standpipe Bubbler was investigated through computerized solution of analytic equations and complemented with testing of a physical model for confidence enhancement of the analytic approach. Sensitivity of pump tank fluid level as a function of: (1) Standpipe Bubbler location, (2) fluid temperature, (3) suction line to tank impedance, (4) cover gas mass flow rate, (5) cover gas system return pressure, and (6) Standpipe Bubbler area and nozzle size is presented in graphical form. Limited testing of the physical model shows close correlation to the computerized model and these data are also presented in graphical form. The preliminary results to date indicate that satisfactory fluid level control can be attained with the Standpipe Bubbler mechanization.

- 2911 (TID--28094) SPTF FLOW LOOP 18 INCH BUTTERFLY VALVE CAVITATION TEST REPORT, SPTF-LPP. Laird, R.A.; Bowers, W.M. (Liquid Metal Engineering Center, Canoga Park, CA (USA)). 29 Jun 1977. Contract EY-76-C-03-0700. 39p. AT.

Portions of document are illegible. Cavitation testing of the HV-130 18 inch butterfly valve located in the Flow Loop at SPTF was conducted from March 22 to April 1, 1977. These tests were performed to determine cavitation characteristics as a function of valve position and upstream pressure in sodium at 600°F. Similar tests had been performed on the same valve using water as the flow medium. Three different test approaches for cavitation were used.

- 2912 (TID--28193) PLBR PROJECT. DESIGN CRITERIA FOR THE PROTOTYPE LARGE BREEDER REACTOR. (Bechtel Corp., San Francisco, CA (USA)). Sep 1977. Contract EY-76-C-03-1144. 444p. AT.

The criteria to be used in the design of the Prototype Large Breeder Report (PLBR) Balance-of-Plant (BOP) are divided into six sections. The first section, General Design Criteria, contains nuclear design criteria, including radiation shielding, safety and NRC licensing requirements. These criteria generally involve more than one engineering discipline. The remaining sections correspond to the following discipline areas: civil/structural, piping stress analysis, mechanical, control systems, and electrical. Architectural design criteria will be included in the Civil/Structural section at a later date. The criteria presented should be regarded as tentative, based on the status of the PLBR design at the conclusion of

Phase II. As the design progresses from the current conceptual state to the detailed design stages, these criteria will be updated as required to meet project needs and any changes that may occur in PLBR project design objectives.

2913 (TID--28194) PROTOTYPE LARGE BREEDER REACTOR: PHASE II EXTENSION. VOLUME I. CONCEPTUAL DESIGN STUDIES. (Bechtel Corp., San Francisco, CA (USA)). Sep 1977. Contract EY-76-C-03-1144. 345p. AT.

The document is a compilation of an extension of the work done during Phase II of the ERDA/Electric Power Research Institute (EPRI) Prototype Large Breeder Reactor (PLBR) study. The objective of this Phase II extension was to investigate key engineering problems generic to the balance-of-plant for the loop concept which were identified during the Phase II effort. Included are the following reports which present the results of these investigations: (1) piping seismic restraint design for intermediate sodium piping between buildings (BTR-9); (2) containment building structural concept study (BTR-11); (3) heat transport system piping reliability enhancement study (BTR-12); (4) minimization of seismic loadings on components (BTR-13); and (5) heat transport cell liner design requirements (BTR-14).

2914 (TID--28195) PROTOTYPE LARGE BREEDER REACTOR. PHASE II EXTENSION. VOLUME II. ELTEMP PROGRAM UPDATE. Chow, H.Y.; Shih, M. (Bechtel Corp., San Francisco, CA (USA)). Sep 1977. Contract EY-76-C-03-1144. 495p. AT.

The ELTEMP Program was developed by Bechtel Power Corporation for design and analysis of Nuclear Class I piping for elevated temperature liquid metal service. This work was done as part of the Fast Flux Test Facility (FFTF) program. The program was originally coded to provide assessments of compliance of Class I piping systems with the stress and strain limits specified in ASME Code Case (C.C.) 1592-3. The program has been used extensively since 1972 in the preliminary design of FFTF and Prototype Large Breeder Reactor Heat Transfer System (PLBR HTS) piping. Since its original development, the program has not been updated to incorporate new developments and changes made in the applicable design codes, e.g., Code Case 1592-10. In the version of ELTEMP described, the creep-fatigue damage evaluation procedure has been completely revised to make it compatible with C.C. 1592-10 requirements, and the materials list expanded to include Incoloy 800 and 2 1/4 Cr-1 Mo Steel. A list of the major modifications made in the current version of ELTEMP Program is provided. The report describes the basic approach, organization, contents, and input data preparation for the new ELTEMP Program. An example of an ELTEMP run is included.

2915 (TID--28198) TURBULENT HYDRAULIC PENETRATION FROM TURBULENT PIPE FLOW INTO STAGNANT COLUMNS. Bloom, G.R. (Hanford Engineering Development Lab., Richland, WA (USA)). Jan 1978. Contract EY-76-C-14-2170. 113p. Dep. NTIS, PC A06/MF A01.

Thesis. Submitted to Washington State Univ., Pullman.

An experimental investigation was conducted on the depth of mixing of water from turbulent pipe flow into a vertically oriented pipe deadleg or stagnant column. The depth of turbulent penetration or mixing length (ML) was correlated as a function of the piping system Reynolds number (Re) and kinematic viscosity

(ν). The dimensionless mixing length (ML) is defined as the depth of hydraulic mixing divided by the inside diameter of the deadleg. Water was pumped at flow rates between 40 l/min and 750 l/min through smooth tubing sections with inside diameters between 22 mm and 73 mm. Isothermal experimental runs were performed at water temperatures between 30°C and 80°C.

2916 (TID--28251) PROTOTYPE LARGE BREEDER REACTOR: PHASE II FOLLOW-ON FINAL REPORT. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Nov 1977. Contract EY-76-C-03-1141. 595p. AT.

The volume includes five separate reports on follow-on tasks which were assigned to Westinghouse Advanced Reactors Division at the end of Phase II of the PLBR studies. Although a decision was made in April 1977, to discontinue the PLBR studies and not to enter Phase III of the Program, the need to investigate a few outstanding issues prior to the end of FY 1977 was identified. Westinghouse ARD was assigned the following tasks: Task I - surveillance and in-service inspection; Task J - piping design and analysis; Task L-1 - primary piping integrity - licensability; Task N - Sulzer cycle - system operation at start-up, shutdown and part-load; and Task O - plant simulation and control of superheat cycle.

2917 (TID--28256) PRIMARY HEAT TRANSPORT SYSTEM AND REACTOR CONTAINMENT BUILDING: LAYOUT SELECTION STUDY. PROTOTYPE LARGE BREEDER REACTOR. PHASE II. Roecker, J.H.; Roe, K.K. (Atomics International Div., Canoga Park, CA (USA); Burns and Roe, Oradell, NJ (USA). Breeder Reactor Div.). Jan 1977. Contract EY-76-C-03-1142; EY-76-C-03-1145. 478p. AT.

The purpose of the study was to develop an optimum integrated primary heat transport system (PHTS) and reactor containment building (RCB) layout. The study basically involved a sequential development approach coordinated around an initial screening, preliminary evaluation and final selection process. Over 14 options were examined and evaluated, including a number of configuration evolutions that developed during the course of the study. Four of these configurations were pursued in detail. During the progression of the design effort, the best features of each group of configurations were carried forward and incorporated into the newer configurations wherever possible. These final four configurations benefitted from these design evolutions. The selected reference PHTS/RCB configuration is characterized by a 170 foot diameter RCB, expansion compensation via expansion loops, and an ex-containment location of the Ex-Vessel Storage Tank (EVST). The configuration has resulted in a PLBR design which is compact, efficient in Reactor Containment Building (RCB) space utilization (all radioactive sodium systems are contained within RCB), and readily licensable. This design has been achieved while retaining all established criteria (e.g., 3 ft. space allowance around piping and components for maintenance and ISI) and maintaining adequate stress design margins to provide contingency for detailed design and analysis.

2918 (TID--28259) PROTOTYPE LARGE BREEDER REACTOR: PHASE II FOLLOW-ON FINAL REPORT. EXECUTIVE SUMMARY. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Nov 1977. Contract EY-76-C-03-1141. 32p. AT.

An Executive Summary is presented of five

reports pertaining to follow-on tasks which were assigned to Westinghouse Advanced Reactors Division at the end of Phase II of the PLBR studies. Although a decision was made in April 1977, to discontinue the PLBR studies and not to enter Phase III of the Program, the need to investigate a few outstanding issues prior to the end of FY 1977 was identified. Westinghouse ARD was assigned the following tasks: Task I--surveillance and in-service inspection; Task J--piping design and analysis; Task L-1--primary piping integrity-licensability; Task N--Sulzer cycle-system operation at start-up, shutdown and part-load; and Task O--plant simulation and control of superheat cycle. In parallel, Stone and Webster was assigned several tasks, for which they are issuing a separate report. A few of the tasks, such as the Surveillance and the Piping Design tasks, required some cooperation between the Westinghouse and the Stone and Webster teams, but most of the work was performed independently. The reports themselves are presented in a separate volume.

- 2919 (TID--28265) PROTOTYPE LARGE BREEDER REACTOR: PRIMARY HEAT TRANSPORT SYSTEM CELL STUDIES. FINAL REPORT, PHASE II EXTENSION. (Stone and Webster Engineering Corp., Boston, MA (USA)). Sep 1977. Contract EY-76-C-03-1146. 236p. AT.

The basic objective of the Prototype Large Breeder Reactor (PLBR) project was to develop the design of an LMFBR power plant that will be licensable and prototypical of subsequent breeder reactor plants. The PLBR is prototypical of all major characteristics of the commercial breeder reactor. The PLBR project was funded jointly by the Energy Research and Development Administration (ERDA) and Electric Power Research Institute (EPRI). Phase I of the program, which lasted 8 months, concentrated on the establishment of basic plant characteristics and the bases for prototype relationship between the PLBR and the commercial breeder reactor. This work was reported in the Phase I Final Report in July 1976. Phase II, which lasted 10 months, was intended to develop the overall plant conceptual design and to select specific concepts for the building arrangements, systems, and major components within the PLBR. Significant technical progress was achieved by the Westinghouse and Stone and Webster team in meeting these objectives concerning the PLBR design during Phase II. This technical progress is documented in detail in 4 volumes issued in June 1977. The technical detail presented in the Phase II report was not intended as an integrated plant design. Additional integrated plant design activities remained to be accomplished, together with detailed evaluation of the plant response to design base events and evaluation of the plant design for events beyond the design base. This integrated plant design had been planned for Phase III of the program which was not pursued.

- 2920 (TID--28269) PROTECTED BAYONET TUBE STEAM GENERATOR PROOF-OF-PRINCIPLE TESTS. FINAL REPORT. Cho, S.M.; Degutis, G.; Govignon, E. (Foster Wheeler Energy Corp., Livingston, NJ (USA)). 30 Sep 1974. Contract EY-76-C-03-0972. 196p. AT.

The results are presented of the sodium-water reaction Proof-of-Principle (POP) testing of the FWC Protected Bayonet Tube (PBT) Steam Generator. Incorporation of the protector tube in the steam generator design has been proven to prevent rapid progression of heat transfer tube failures resulting from a leaking water/steam tube, and thus increases the time available for an orderly shutdown of the steam

generator by an order of magnitude. The test program was performed in two test rigs; the SOWAT/SOLLACE and the Protector Tube Test Rig (PTTR). The test program performed in the PTTR provided full-size cross-section and full length protector tube test bundles. The water injection rate increased from 0.6×10^{-2} lb/sec up to 10 lb/sec, with the maximum total mass of 20 lb. The results of all tests are included in this report.

- 2921 (TID--28273/1) FOSTER WHEELER PROTECTED BAYONET TUBE LMFBR STEAM GENERATOR PRELIMINARY DESIGN. Govignon, E.; Wolowodiuk, W. (Foster Wheeler Energy Corp., Livingston, NJ (USA)). 1974. Contract EY-76-C-03-0972. 254p. AT.

Portions of document are illegible.

The preliminary design report presents the work accomplished by Foster Wheeler during the period from March 14, 1974 to August 23, 1974 on the PBT LMFBR Steam Generator Preliminary Design. This work was a follow on effort to the work reported in FWC document ND-73-69 dated December 21, 1973 and entitled "Conceptual Design LMFBR Bayonet Tube--Protector Tube Steam Generator." The PBT Design has evolved from earlier work conducted on the ESADA 7 tube evaporator work and on the PBT Proof of Principle testing as reported in FWC ND-74-58. General Electric Company Nuclear Energy Division has served as technical monitor for the effort which has been supported by the Atomic Energy Commission under AEC Contract AT(04-3)-972.

- 2922 (TID--28273/2) FOSTER WHEELER PROTECTED BAYONET TUBE LMFBR STEAM GENERATOR PRELIMINARY DESIGN. VOLUME II. APPENDICES. (Foster Wheeler Energy Corp., Livingston, NJ (USA)). 1974. Contract EY-76-C-03-0972. 291p. Dep. NTIS, MF A01.

Portions of document are illegible.

The appendix contains the sizing calculations for the major components of the steam generator. All calculations were performed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 1, with addenda through the Winter of 1973 and Code Case 1331-8.

- 2923 (TID--28546) SUITABILITY OF 2--1/4CR--1MO STEEL FOR LMFBR STEAM GENERATION SYSTEM APPLICATIONS: A CRITIQUE OF NUREG-0387. Spalaris, C.N. (comp.). (General Electric Co., San Jose, CA (USA). Fast Breeder Reactor Dept.). 1 May 1978. Contract EY-76-C-15-1200. 59p. AT.

Responses of industry representatives to a report which questioned the suitability of 2 1/4 Cr-1 Mo steel for LMFBR steam generators are presented. The responses reendorse the choice of 2 1/4 Cr-1 Mo steel for steam generator applications.

- 2924 (TID--28710) AI SELECTION OF SINGLE WALL OVER DOUBLE WALL LMFBR STEAM GENERATORS. (Atomic International Div., Canoga Park, CA (USA)). 1978. Contract EY-76-C-03-1142. 9p. AT.

The decision by AI to employ a single wall steam generator was based upon consideration of the following factors: (1) reliability and experience vis-a-vis single versus double wall designs, (2) technical aspects of the single versus double wall units, and (3) economic aspects of single versus double wall steam generators.

2925 (TID--28712) B AND W BASIS FOR DESIGN SELECTIONS FOR INTERMEDIATE "A" SINGLE WALL STEAM GENERATOR DESIGN. (Babcock and Wilcox Co., Barberton, OH (USA)). [nd]. Contract EN-77-C-02-4176. 9p. AT.

A design description of a single wall LMFR steam generator is presented. The basis for design selections is described.

2926 (TID--28713) DEVELOPMENT AND TEST RESULTS OF WESTINGHOUSE DOUBLE WALL TUBE STEAM GENERATOR INCLUDING THERMAL MODEL TEST. Lohmeier, A.; Bongaards, D.J. (Westinghouse Electric Corp., Tampa, FL (USA). Tampa Div.). Dec 1974. Contract EY-76-C-03-0962. 22p. AT.

Recognizing consequences of sodium water reaction in sodium heated steam generators, a study program at Westinghouse Tampa Division demonstrated design feasibility of utilizing duplex tubes with helium interface fluid to provide for sodium and water tube leak detection such that probability of sodium-water reaction is reduced. Program elements demonstrated grooved duplex tube manufacturing and inspection methods, thermal performance predictability and small sodium or water leak detection. A prototypic single-tube J-module steam generator model test was utilized to demonstrate thermal-hydraulic performance of the duplex tube under prototypic plant operating conditions.

2927 (TID--28715) BASIS FOR WESTINGHOUSE DOUBLE WALL TUBE STEAM GENERATOR WITH THIRD FLUID LEAK DETECTION CAPABILITY EXTRACTED FROM WESTINGHOUSE PROPOSAL TO THE DEPARTMENT OF ENERGY TO DEVELOP A STEAM GENERATOR FOR LARGE SCALE LIQUID METAL FAST BREEDER REACTORS. (Westinghouse Electric Corp., Tampa, FL (USA). Tampa Div.). Jun 1976. Contract EN-77-C-02-4177. 105p. AT.

Westinghouse is concerned about the potential consequences of having a sodium/water reaction in an operating sodium heated steam generator unit. This concern stems from an assessment of possible damage mechanisms resulting from the distribution of caustic reaction products throughout the system and the associated lengthy downtimes that appear required for cleanup, repair and recommissioning. It is proposed that a double-wall tube steam generator with third fluid leak detection capability be utilized in the first commercial LMFR plants. More specifically, the proposed high and low temperature commercial plant steam generators should be maximum size (350 MWT and 600 MWT, respectively) straight shell and tube integral evaporator and superheater once-through heat exchangers (as opposed to serpentine or helical coil tube units) to emphasize economic considerations without loss of component reliability. The proposed steam generator concepts are described.

2928 (TID--28746) BOUYANCY EFFECTS ON SODIUM COOLANT TEMPERATURE PROFILES MEASURED IN AN ELECTRICALLY HEATED MOCK-UP OF A 61-ROD BREEDER REACTOR BLANKET ASSEMBLY. Engel, F.C.; Markley, R.A.; Minushkin, B. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1978. Contract EY-76-C-02-3045-004. 20p. Dep. NTIS, PC A02/MF A01.

The paper describes test results selected to demonstrate the effect of buoyancy on the temperature profiles in a 61-rod electrically heated mock-up of an LMFR radial blanket assembly. In these assemblies, heat transfer occurs over a wide range of complex operating conditions. The range and complexity of

conditions are the result of the steep flux and power gradients which are an inherent feature of the blanket region and the power generation level in an assembly which can vary from 20 to 1100 kW.

2929 (TID--28755) FINAL REPORT: PHASE II EXTENSION, PROTOTYPE LARGE BREEDER REACTOR. PRIMARY HEAT TRANSPORT SYSTEM CELL STUDIES. (Stone and Webster Engineering Corp., Boston, MA (USA)). Sep 1977. Contract EY-76-C-03-1146. 229p. AT.

As a result of the premature suspension of the PLBR design activities, a number of questionable areas remained outstanding regarding the feasibility and practicality of the loop plant design. ERDA desired that these areas related to the feasibility of the loop plant design be studied in some detail during a 4 month period ending on September 30, 1977. Accordingly, Stone and Webster studied certain aspects of the PHTS cell design in more detail during this Phase II extension activity. The following objectives were established for this effort. TASK I--PHTS CELL DESIGN: Evaluate the primary heat transport system (PHTS) cell design features for a range of sodium spills. Establish PHTS cell design features that will contain sodium spills, limit sodium-concrete reactions, prevent excessive pressure buildup, and accommodate other effects of sodium leakage events. The cell configuration established at the end of Phase II will be used for this study. TASK II--PRIMARY PIPING DESIGN AND INSPECTION: Verify acceptability of Phase II primary piping arrangement for seismic conditions, determine location of pipe hangers and seismic snubbers, and define attachments to the cell structure. Evaluate methods of monitoring hanger and snubber performance and the capabilities to perform in-service inspection of the hangers and snubbers. TASK III--PHTS CELL PROVISIONS FOR MAINTENANCE: Investigate and evaluate the Phase II PHTS cell arrangement and features to facilitate manned access for local maintenance of primary components and recommend appropriate design changes. The objectives were met by design and analytical studies of PHTS cell structures, cell liners and sodium catch pans, primary piping supports, maintenance approaches, and by preparation of detailed composite drawings and construction of a scale model of the PHTS cell.

2930 (TID--28756) PROTOTYPE LARGE BREEDER REACTOR. PHASE II. FOLLOW-ON FINAL REPORT. Anderson, C.A. Jr. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Nov 1977. Contract EY-76-C-03-1141. 573p. AT.

The volume includes five separate reports on tasks performed for the Department of Energy (previously ERDA, RDD) during the period of June 1, 1977 to October 1, 1977. These tasks are follow-on tasks which were assigned to Westinghouse Advanced Reactors Division at the end of Phase II of the PLBR studies. Although a decision was made in April 1977, to discontinue the PLBR studies and not to enter Phase III of the Program, the need to investigate a few outstanding issues prior to the end of FY 1977 was identified. Westinghouse ARD was assigned the following tasks: (1) surveillance and in-service inspection; (2) piping design and analysis; (3) primary piping integrity-licensability; (4) suizer cycle--system operation at start-up, shutdown and part-load; and (5) plant simulation and control of superheat cycle.

2931 (TID--28758) PROTOTYPE LARGE BREEDER

REACTOR: PHASE II EXTENSION. VOLUME I. CONCEPTUAL DESIGN STUDIES. (Bechtel Corp., San Francisco, CA (USA)). Sep 1977. Contract EY-76-C-03-1144. 411p. AT.

The document is a compilation of work sponsored by the Energy Research and Development Administration (ERDA) as an extension of the work done during Phase II of the ERDA/Electric Power Research Institute (EPRI) Prototype Large Breeder Reactor (PLBR) study. The objective of this Phase II extension was to investigate key engineering problems generic to the balance-of-plant for the loop concept which were identified during the Phase II effort. Volume I presents the results of the following investigations: (1) piping seismic restraint design for intermediate sodium piping between buildings (BTR-9), (2) containment building structural concept study (BTR-11), (3) heat transport system piping reliability enhancement study (BTR-12), (4) minimization of seismic loadings on components (BTR-13), and (5) heat transport cell liner design requirements (BTR-14).

- 2932 (TID--28937) REVIEW OF LITERATURE ON UNSTEADY FORCED CONVECTION HEAT TRANSFER WITH APPLICATION TO LMFBR THERMAL STRIPING. Snyder, P.H. (Westinghouse Electric Corp., Madison, PA (USA)). Advanced Reactors Div.). Jan 1979. Contract EY-76-C-15-2395; EY-78-C-15-0003. 47p. AT.

A literature review was conducted to determine what techniques are available to predict the unsteady forced convection heat transfer behavior of LMFBR upper core parts. The most suitable techniques are explored and pertinent methods are evaluated and extended to analyze the problem at hand. It is determined that available methods employ assumptions too restrictive to analyze the required situation and that further work should be aimed at the extension of these methods.

- 2933 (TID--29052) NUFACFS--NUCLEAR FUEL CYCLE ACTIVITY SIMULATOR: REFERENCE MANUAL. FINAL REPORT. Triplett, M.B.; Waddell, J.D.; Breese, T.A. (Battelle Columbus Labs., OH (USA)). 15 Aug 1978. Contract W-7405-ENG-92-089. 238p. Dep. NTIS, PC A11/MF A01.

Portions of document are illegible.

The Nuclear Fuel Cycle Activity Simulator (NUFACFS) is a package of Fortran subroutines which facilitate the simulation of a diversity of nuclear power growth scenarios. An approach to modeling the nuclear fuel cycle has been developed that is highly adaptive and capable of addressing a variety of problems. Being a simulation model rather than an optimization model, NUFACFS mimics the events and processes that are characteristic of the nuclear fuel cycle. This approach enables the model user to grasp the modeling approach rather quickly. Descriptions of the model and its components are provided with several emphases. First, a discussion of the modeling approach and basic assumptions is provided. Next, instructions are provided for generating data, inputting the data properly, and running the code. Finally, detailed descriptions of individual program element are given as an aid to modifying and extending the present capabilities.

- 2934 (TID--29088) NUCLEAR POWER WITHOUT NUCLEAR BOMBS. LINEAR ACCELERATOR-BREEDER CONCEPT FOR DECOUPLING THE NUCLEAR POWER ECONOMY FROM NUCLEAR WEAPONS PRODUCTION. Steinberg, M. (Brookhaven National Lab., Upton, NY (USA)). Jan 1977. Contract EY-76-C-02-0016. 15p. Dep. NTIS, PC A02/MF A01.

The role of nuclear weapon technology in

starting today's nuclear power plants is pointed out. There is another technology that can give us nuclear power without proliferation of weapons production: the linear accelerator-breeder (LAB) in which high-energy protons impinge on heavy metal targets to produce many neutrons which are absorbed in natural fertile material to produce fissile fuel for fission power reactors. Waste can also be transmitted in the same way. This is accomplished without the need for enrichment or reprocessing. The advantages of the LAB over LMFBRs are clear. The fuel cycle using the LAB device is described. Reprocessing and enrichment can only increase the supply of fissile fuel up to 30% whereas LAB can burn up practically all the fertile material which is an increase of 14,000%. (DLC)

- 2935 (TID--29350) DISTRIBUTION OF REVISIONS TO FFTF FSAR. Schade, A.R. (Hanford Engineering Development Lab., Richland, WA (USA)). 9 Nov 1978. Contract EY-76-C-14-2170. 137p. AT.

An amendment dealing with environmental design of mechanical and electrical equipment is presented for inclusion in the FFTF Final Safety Analysis Report

- 2936 (TR--097-330-009) MODULAR STEAM GENERATOR TEST REPORT: POST-PERFORMANCE TEST DISASSEMBLY AND EXAMINATION. Friske, W.H.; Page, J.P.; Williams, T.D. (Atomics International, Canoga Park, CA (USA)). 30 Aug 1974. Contract EY-76-C-03-0824-014. 41p. AT.

Selected regions of the Atomics International Modular Steam Generator (AI-MSG) were examined for the effect of 4000 hr of steaming service. Visual, dimensional, metallographic, and chemical techniques were employed in a search for: (1) water-steam-side effects, such as fouling, erosion, corrosion, and exfoliation, (2) sodium-side effects, such as corrosion, carbon transport, wear, fretting, erosion, and fouling, (3) overall effects, such as evidence of creep, fatigue, any evidence of infant mortality failure modes, and deterioration of properties of the wrought and welded 2-1/4 Cr - 1 Mo steel of which the steam generator is constructed, or (4) mechanical interference. On the basis of the results of this examination, it is concluded that both the basic design and the materials of construction are adequate for long-term LMFBR service.

- 2937 (TR--097-330-010) MODULAR STEAM GENERATOR FINAL PROJECT REPORT. Harty, R.B. (Atomics International Div., Canoga Park, CA (USA)). Sep 1974. Contract EY-76-C-03-0824-014. 206p. Participants in the LMFBR Program. Others request from TIC.

Atomics International designed, fabricated, and tested a prototype sodium cooled steam generator unit termed the Modular Steam Generator (MSG). The overall test program was divided into different phases, each with the intent of providing specific operational and performance information. These tests include preheat, vibration, pressurization (strain gauge), startup/shutdown, steady-state operation and heat transfer (thermal/hydraulics), impurities injection (leak detection), low-flow stability, and transients. Highlights of this test program are given. Test conditions at both a nominal 2400 and 1450 psig steam turbine pressure were tested at temperatures up to 950°F. Following the test program, a post-test disassembly and examination was conducted to determine if there was any loss of integrity or other potential

failure modes as a result of the imposed service conditions. The findings of this post-test examination are summarized. No evidence of any deleterious effects were found. Although the total operating time in the Sodium Components Test Installation represented but a few percent of the planned operating life in an LMFBR operating plant, the number of tube vibration cycles (approximately 10^9 cycles), localized temperature oscillations (10^7 on sodium-side lower mixing region, 10^8 in tube-side DNB region), and velocities in the bend region were such as to expose any evidence of possible wearout modes. Therefore, it was concluded that results provide strong indication of design suitability for long term service.

- 2938 (TREE--1116, pp 128-130) EVALUATED DECAY-SCHEME DATA FOR ILRR. Helmer, R.G.; Greenwood, R.C. Apr 1977.

Nuclear Technology Division annual progress report for period ending September 30, 1976.

The Interlaboratory LMFBR Reaction Rate (ILRR) program was organized for the purpose of making a series of precise measurements of rates of neutron-induced reactions for a set of well-defined and reproducible neutron spectra. For many of these reactions, the rates are determined by measurements of the gamma-ray spectra from the decay of the radioactive reaction products. A goal of the ILRR program is to measure fission reaction rates by means of gamma-ray spectrometry with accuracies of about ± 5 percent at the 95 percent confidence level. The computation of the reaction rates depends on two parameters of the nuclear decay scheme; these are the half-life and the absolute gamma-ray intensity. For the ILRR program, it is necessary that these parameters and their errors be as accurate and consistent as possible. Therefore, a series of regular reviews of the decay schemes of interest has been undertaken. Two earlier reviews were noted in Reference 1-3. The results of the most recent evaluation are given in Table I for selected gamma-rays. Original references, and published evaluations available as of November 1975 are included.

- 2939 (TREE--1168) USER'S GUIDE FOR THE RELIABILITY ANALYSIS SYSTEM (RAS). Rasmuson, D.M.; Marshall, N.H.; Burdick, G.R. (Idaho National Engineering Lab., Idaho Falls (USA)). Sep 1977. Contract EY-76-C-07-1570. 117p. Dep. NTIS, PC A06/MF A01.

The Reliability Analysis System (RAS) is an integrated package of computer programs for the quantification of fault trees. In particular, RAS has the capacity of performing phased mission analysis. RAS is based upon the MOCUS, POCUS, KITT-1, SRTPRN, and COMCAN programs. Mission cut set cancellation and four methods of bounding mission unreliability have been incorporated into RAS. A new cut set algorithm, FATRAM, is also included in RAS. RAS is written for the CDC Cyber-76 computer system and utilizes external storage devices. The program is available from the Argonne Code Center, 9700 South Cass Avenue, Argonne, Illinois 60439.

- 2940 (TREE--1255) EVALUATION OF ROLL-EXTRUDED ALLOY 718 TUBING. Smolik, G.R.; Korth, G.E. (Idaho National Engineering Lab., Idaho Falls (USA)). May 1978. Contract EY-76-C-07-1570. 44p. Dep. NTIS, PC A03/MF A01.

A sample of roll-extruded Alloy 718 tubing, a product identified for use in the Clinch River Breeder Reactor, was evaluated. The tubing satisfied AMS 5589 requirements for

seamless tubing, and had tensile and fatigue properties comparable to plate for the conventional 955°C (1750°F) heat treatment. Recrystallization was retarded in this product and bands of very small grains resulted from the conventional heat treatment. Both as-roll-extruded and directly aged material showed the results of thermomechanical processing in high strengths and reductions in area. It is recommended that work continue toward realizing improved fatigue and stress rupture properties in this product by means of 'Minigrain' and thermomechanical processing. 10 figures, 6 tables.

- 2941 (TREE--1289) COMCAN II: A COMPUTER PROGRAM FOR COMMON CAUSE FAILURE ANALYSIS. Rasmuson, D.M.; Wilson, J.R.; Marshall, N.H.; Burdick, G.R. (Idaho National Engineering Lab., Idaho Falls (USA)). Sep 1978. Contract EY-76-C-07-1570. 76p. Dep. NTIS, PC A05/MF A01.

This report is the user's manual for COMCAN II, a computer program for identifying possible common causes for the failure of fault tree minimal cut sets. The program is written in FORTRAN IV for the CDC CYBER-76 computer. A main feature of the program is its ability to identify minimal cut sets of any order with failure potential due to a single secondary failure (common cause).

- 2942 (TREE--1360) EVALUATION OF ROLL-EXTRUDED ALLOY 718 TUBES: SEAMLESS VERSUS WELDED. Smolik, G.R.; Flinn, J.E.; Korth, G.E. (Idaho National Engineering Lab., Idaho Falls (USA)). May 1979. Contract EY-76-C-07-1570. 77p. AT.

Large-diameter tubes of Alloy 718 are required for the upper core chimneys for the Clinch River Breeder Reactor and other liquid metal fast breeder reactor applications. This investigation was performed to: (1) determine the feasibility of using welded cylindrical blanks for roll-extrusion stock and (2) obtain material property data from roll-extruded tubes. Seamless and welded tubes were roll extruded from a common material source, Alloy 718 reference heat. Recrystallization, tensile, creep-rupture and fatigue behavior were determined for both tubes. Effects of solution anneal temperature upon mechanical properties of the seamless tube were checked. The effect of orientation upon selected properties of both the seamless and welded tubes was also examined. Finally, tensile properties of the roll-extruded tubes were compared with AMS specification requirements and data for cylindrical products formed by other processes.

- 2943 (TREE--1361) COMCAN II-A: A COMPUTER PROGRAM FOR AUTOMATED COMMON CAUSE FAILURE ANALYSIS. Rasmuson, D.M.; Marshall, N.H.; Wilson, J.R.; Burdick, G.R. (Idaho National Engineering Lab., Idaho Falls (USA)). May 1979. Contract EY-76-C-07-1570. 84p. Dep. NTIS, PC A05/MF A01.

The user's manual for COMCAN II-A, a computer program that identifies possible common causes for the failure of fault tree minimal cut sets, is presented. The program also contains options to help an analyst construct common locations and to assign susceptibilities to the components. The program is written in FORTRAN IV for the CDC CYBER-76 computer. A main feature of the program is its ability to identify minimal cut sets of any order with failure potential due to a single secondary failure (common cause).

2944 (UAC--41069) CONTROL OF OXYGEN, HYDROGEN, AND TRITIUM IN SODIUM SYSTEMS AT EXPERIMENTAL BREEDER REACTOR II. Osterhout, M.M. (Argonne National Lab., IL (USA)). 1978. Contract W-31-109-ENG-38. 14p. Dep. NTIS, PC A02/MF A01.

The source rates and steady-state concentrations have been measured for oxygen, hydrogen, and tritium impurities in EBR-II sodium systems. An analytical approach is used to determine the effectiveness of the cold traps for removing oxygen, hydrogen, and tritium. The cold-trap effectiveness data accumulated to date for removal of oxygen, hydrogen, and tritium indicate that EBR-II cold traps are highly effective in controlling these impurities to very low concentrations in the sodium systems.

2945 (UCRL--13916(Pt.1)(Add.)) MATERIAL CONTROL IN NUCLEAR FUEL FABRICATION FACILITIES. ADDENDUM TO PART I: FUEL DESCRIPTIONS AND FABRICATION PROCESSES. Borgonovi, G.M.; McCartin, T.J.; Miller, C.L.; Nguyen, T. (Science Applications, Inc., McLean, VA (USA)). Oct 1978. Contract W-7405-ENG-48. 27p. Dep. NTIS, PC A03/MF A01.

The addendum contains additional information on existing and planned nuclear fuel fabrication facilities in the world, and descriptions of a selected number of facilities including a conceptual facility which replaces the one initially presented and has been used as a basis for the discussions of material accountability and control. 8 figures, 4 tables.

2946 (UCRL--13942(Pt.1)) MATERIAL CONTROL IN NUCLEAR FUEL FABRICATION FACILITIES. PART I. FUEL DESCRIPTIONS AND FABRICATION PROCESSES, P.O. 1236909 FINAL REPORT. Borgonovi, G.M.; McCartin, T.J.; Miller, C.L. (Science Applications, Inc., McLean, VA (USA)). Dec 1978. Contract W-7405-ENG-48. 75p. (SAI--79-88-WA).

The report presents information on foreign nuclear fuel fabrication facilities. Fuel descriptions and fuel fabrication information for three basic reactor types are presented: The information presented for LWRs assumes that Pu-U Mixed Oxide Fuel (MOX) will be used as fuel.

2947 (UCRL--79653(Rev.1)) LASER DRIVEN FUSION FISSION HYBRIDS. Hansen, L.F.; Maniscalco, J.A. (California Univ., Livermore (USA). Lawrence Livermore Lab.). Nov 1977. Contract W-7405-ENG-48. 38p. (CONF-771102--15). Dep. NTIS, PC A03/MF A01.

From 20. annual AIChE meeting; New York, NY, USA (13 Nov 1977).

The role of the fusion-fission hybrid reactor (FFHR) as a fissile fuel and/or power producer is discussed. As long range options to supply the world energy needs, hybrid-fueled thermal-burner reactors are compared to liquid metal fast breeder reactors (LMFBR). A discussion of different fuel cycles (thorium, depleted uranium, and spent fuel) is presented in order to compare the energy multiplication, the production of fissile fuel, the laser efficiency and pellet gain requirements of the hybrid reactor. Lawrence Livermore Laboratory (LLL) has collaborated with Bechtel Corporation and with Westinghouse in two engineering design studies of laser fusion driven hybrid power plants. The hybrid designs which have resulted from these two studies are briefly described and analyzed by considering operational parameters, such as energy multiplication, power density, burn-up and plutonium production

as a function time.

2948 (UCRL--82614(Rev.1)) ELASTIC AND PLASTIC PROPERTIES OF URANIUM DIOXIDE FROM 5 TO 330 GPa. Gust, W.H. (California Univ., Livermore (USA). Lawrence Livermore Lab.). 11 Jul 1979. Contract W-7405-ENG-48. 5p. (CONF-790709--3(Rev.1)). Dep. NTIS, PC A02/MF A01.

From VII international AIRAPT conference; La Creusot, France (30 Jul 1979).

We have measured the shock-compression parameters for UO_2 to 330 GPa. The Hugoniot elastic limit was found to be 5.7 GPa. Evidence for a shock-induced phase transition was observed at about 54 GPa.

2949 (UJV--4156-T) RESULTS OF MEASURING LOCAL HYDRODYNAMIC CHARACTERISTICS OF GEOMETRICALLY DISTURBED ROD BUNDLE. I. Hejna, J.; Cervenka, J.; Mantlik, F. (Ustav Jaderneho Vyzkumu CSKA, Rez (Czechoslovakia)). Oct 1977. 68p. (In Russian). Dep. NTIS (US Sales Only), PC A04/MF A01.

The local hydrodynamic characteristics were studied on an open aerodynamic loop with a 19-rod bundle modelling the coolant flow through the fuel element of the BN-600 fast reactor. The relative lattice pitch in the bundle was $s/d=1.17$, the outer diameter of the model was 120 mm and total length 6,000 mm. The measuring cross-section was at a distance of 5,600 mm from the inlet. The central rod was deflected in the direction of the neighbouring rods with excentricities of 15.0 mm and 20.4 mm. The flow through the model was within the Reynolds number region of 47,800 to 141,000. Measured was the velocity field and the course of shear stress along the wetted perimeter. The results are given in tables and graphs. Friction losses were obtained as secondary results of the measurements which showed that the shift of the central rod had negligible effect on the losses and was comparable with errors in measurement.

2950 (UJV--4333-T) HEAT TRANSFER IN THE PERIPHERAL REGION OF A FAST REACTOR FUEL ASSEMBLY. PART I. EXPERIMENTAL EQUIPMENT; SOLUTION PREREQUISITES, TECHNOLOGICAL EXPERIMENTS. Sulc, V.; Masek, V. (Ustav Jaderneho Vyzkumu CSKA, Rez (Czechoslovakia)). May 1978. 52p. (In Russian). Dep. NTIS (US Sales Only), PC A04/MF A01.

Sodium stand S III was reconstructed at UJV Rez for the research of the effect of radial displacement of a side rod on the temperature field in a BN-600 fast reactor fuel assembly. A model was built of the BN-600 type fast reactor fuel assembly designated EK-16 containing 14 heating and 2 measuring rods. In heated rods the maximum linear heat flow of 53 W/cm² was attained on the outer rod surface. The differential equation is solved of the temperature field on a heating rod consisting of a heating coil, electric insulation and an outer tube. The course of the mean temperature is shown graphically in the rod and the coolant along the EK-16 channel at a power of 1 kW. Also shown graphically are the hydrodynamic and heat characteristics of stand S III with the EK-16 measuring section. The location of NiCr-Ni microthermocouples on measuring rods is described. Vacuum distillation was used in measuring the sodium oxide content. Technological experiments confirmed that the reconstructed stand can be used for the research of temperature fields in BN-600 type reactors with desired accuracy.

2951 (UJV--4576-T) EFFECT OF CONFIGURATION

DISTORTION ON VELOCITY AND TEMPERATURE FIELDS IN THE PERIPHERAL ZONE OF A FAST REACTOR FUEL ASSEMBLY AT LAMINAR FLOW. Milbauer, P. (Ustav Jaderneho Vyzkumu CSKAE, Rez (Czechoslovakia)). Nov 1978. 25p. (In Russian). NTIS (US Sales Only), PC A03/MF A01.

Calculations were performed of hydrodynamic and thermal characteristics for the peripheral zones of the BN-600 reactor fuel assembly. The distributions are shown of the tangential stress coefficient and of dimensionless thermal inhomogeneities along the fuel rod peripheries. The fundamental equations used in the calculations are reported.

2952 (UTNL-R--0030) EXPERIMENTAL STUDY ON FAST NEUTRON STREAMING THROUGH GRID-PLATE SHIELD OF A LMFBR. Oka, Y.; Wakabayashi, H.; An, S.; Suzuki, I. (Tokyo Univ. (Japan). Nuclear Engineering Research Lab.). 20 Jul 1976. 30p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Neutron streaming through the holes penetrating the grid plate shield of a prototype LMFBR was experimentally examined. The mockups of the grid plate shield were made of iron and aluminum. Experiments were conducted at the vertical column of 'YAYOI', the fast neutron source reactor of University of Tokyo. A He-3 spectrometer was employed in order to measure the transmitted neutron spectrum, while rhodium and indium threshold foils were for the integral flux above specific energies and their spatial distributions in the form of reaction rates. The streaming factor for usual small banded holes is 1.28 ± 0.04 as to the integral neutron flux above 0.1 MeV and 1.30 ± 0.12 as to the reaction rate of indium foil. Use were made of the one and two dimensional neutron transport code ANISN and TWOTRAN for evaluation by computation. The reaction rates calculated by infinite slab model with ANISN code agree well with the experiments when normalized at the source point where neutrons are incident on the grid plate shield.

2953 (UTNL-R--0046, pp 44-49) TWO-DIMENSIONAL SHIELDING BENCHMARKS FOR SODIUM AND IRON AT YAYOI. Oka, Y. (Tokyo Univ. (Japan). Nuclear Engineering Research Lab.). Jun 1977. (In Japanese).

From Meeting on heat and structures in nuclear engineering, and symposium on numerical analysis of transport and diffusion phenomena in nuclear engineering; Tokai, Ibaraki, Japan (16 Mar 1977).

The benchmark experiments on two-dimensional shielding for sodium and iron with the reactor YAYOI were performed to obtain the data for designing fast breeder reactors. The deep (transparent) experiment with a plane source is described. The experiments were made with an iron sample of 94 cm x 94 cm x 70 cm, and with a sodium sample of 1 m x 1 m x 1.8 m. The distribution and attenuation of neutrons and gamma-ray in the samples were measured by activation foils and TLD. The fast neutron spectra at the injection points of the samples were observed with threshold detectors. The sandwich foil method was applied for the resonance energy region. Calculation was made by using the one dimensional and two dimensional discrete ordinate transport code ANISN and TWOTRAN-second. The experiment and calculated values on the central axis of the samples were compared, and the agreement between both values is not very good for the attenuation, whereas it is good for the shape of spectra.

2954 (WARD--353) RADIAL PARFAIT CORE DESIGN STUDY. Paxson, E. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jun 1977. Contract EY-76-C-03-1141. 205p. AT.

A study to design a core of the radial parfait (or bullseye) type for PLBR which, through its inherent characteristics, would give assurance that energetic core disruption would not result from transient overpower (TOP) or transient undercooling (TUC) accidents without scram, is described. In addition to the freedom from core disruption, the resulting core design was evaluated with respect to its effect on other reactor characteristics and components such as: thermal hydraulic performance, fuel and blanket assembly mechanical designs and performance, reactor structural design, refueling operation and scheduling, and fuel cycle and capital costs. During the course of the study, two refueling options developed, namely batch refueling on a multi-year cycle and partial core refueling on an annual cycle. Therefore, an additional objective was to make a recommendation relative to these two refueling approaches.

2955 (WARD-AD--3045-6) THEORY OF PARTICLE REDISTRIBUTION IN AN IRRADIATION ENVIRONMENT. Baron, M.; Chang, A.; Bleisberg, M.L. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Apr 1977. Contract EY-76-C-02-3045-024. 25p. AT.

A new theory of particle redistribution in an irradiation environment is proposed based on the physical ideas of Nelson et al. The theory is a natural extension of the Lifshitz-Slyozov-Wagner approach to thermal coarsening. An equation governing the kinetic evolution of the particle size distribution is derived. It is solved for the equilibrium (i.e., displacement dose independent) distribution and together with the conservation of solute, provides a theoretical characterization of the equilibrium state. It is shown that due to solute resolution, the equilibrium state is characterized by the existence of a maximum stable radius and an amount of solute in dynamic solution which is in excess of the equilibrium thermal concentration for the irradiation temperature. Explicit formulae are given for the shape of the universal equilibrium particle size distribution, the maximum and average particle size at equilibrium, the excess amount of solute in dynamic solution and the concentration of particles per unit volume. The asymptotic form of the time dependent particle size distribution is discussed using the dynamic scaling hypothesis. The kinetic development of the maximum and average particle sizes are calculated based on dynamic scaling. Various special cases of the kinetic law governing the relaxation kinetics of the maximum (or average) particle size to their equilibrium values are discussed. Experimental data which are in excellent quantitative and qualitative agreement with theoretical predictions are presented.

2956 (WARD-CG--3045-2) SODIUM/CAVITY GAS EFFECT ON FATIGUE CRACK GROWTH IN 304SS UNDER FFTF PRIMARY COLD LEG PIPING CONDITIONS. Gowda, B.; Matheys, L. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Mar 1978. Contract EY-76-C-02-3045-035. 46p. AT.

This report covers a portion of the work performed to determine the effect of sodium/cavity gas reaction products on fatigue crack growth in Type 304SS pipes. This program was

initiated to answer an important pipe integrity question, i.e. what is the effect of sodium/cavity gas reaction products on the possible extension of a fatigue crack in LMFBR primary piping. Following a description of the test apparatus and the specimens, the results are given for the first six of ten tests scheduled for Type 304SS. It is shown that the fatigue crack growth rate for Type 304SS in the sodium leak environment is not significantly different from that for a pure sodium environment at similar temperatures (600°F to 800°F) and stresses. It is also observed that the geometry of the specimens used (pipes) did not significantly alter the crack growth rates when compared to HEDL crack growth results obtained from compact tension type fracture specimens under conditions similar to the present tests.

- 2957 (WARD-CG--3045-3) CRACK GROWTH IN TYPES 304 AND 316 STAINLESS STEEL UNDER PROTOTYPIC LMFBR PRIMARY PIPING CONDITIONS. Gowda, B.C.; Matheys, W.L. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1978. Contract EY-76-C-3045-035. 31p. AT.

The report presents test operations and results for tests 7 through 15 of the Sodium Effect on Crack Growth (SECG) program. The SECG program was initiated to determine whether the growth rate of a through-wall crack in LMFBR primary piping was affected by sodium leaking through the crack into the cavity gas. Tests to date (including tests 1-6 reported earlier) indicate that fatigue crack growth in both types of 304 SS and 316 SS is inhibited by a sodium/cavity gas environment to the same extent as that in pure sodium. From the limited number of tests it was observed that changes within the range of test variables (1 to 2 percent oxygen and 1000 to 5000 vppm moisture in cavity gas, 400 to 1015°F specimen temperature, 17.23 and 24.10 ksi of maximum stress) did not alter the benign nature of the sodium cavity gas environment. Tensile hold-time seems to slightly increase crack growth in type 304 SS pipes whereas it does not affect crack growth in 316 SS at the conditions and cycle frequency used in testing. However, the increase in crack growth rate with hold-time is not very significant and also is not consistent with some in-air test program results obtained at other laboratories.

- 2958 (WARD-CG--3045-4) ELEVATED TEMPERATURE FATIGUE CRACK GROWTH BEHAVIOR OF TYPE 304/308 STAINLESS STEEL WELDMENTS IN SODIUM/CAVITY GAS ENVIRONMENT. Gowda, B.C.; Matheys, W.L. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Mar 1979. Contract EY-76-C-02-3045-035. 35p. AT.

Results of four tests on Type 304 Stainless Steel pipes with Type 308SS weldments performed under the Sodium/Cavity Gas Effect on crack growth program are presented. The objective of the program was to determine whether the growth rate of a through wall crack in LMFBR primary piping is affected by sodium leaking through the crack into the cavity gas or by interactions between crack tip material and sodium/cavity gas reaction products. Tests indicate that crack growth rates in welds and in the heat affected zone (HAZ) are decelerated by the presence of sodium/cavity gas environment present in the crack vicinity. Two tests conducted at 400°F showed faster crack growth rates than the two tests at 800°F. This observation is tentatively attributed to the higher stress levels used in tests at 400°F.

- 2959 (WARD-CN--3045-4) ANALYTICAL

COMPARISON OF THE IRRADIATION BEHAVIOR OF FAST REACTOR CARBIDE AND OXIDE FUEL PINS. Nayak, U.P.; Boltax, A.; Skalka, R.J.; Blancheria, A. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1977. Contract EY-76-C-02-3045-027. 38p. AT.

Advanced breeder reactor system design studies have shown that mixed U,Pu carbide fuel provides significant advantages over U,Pu oxide fuel with respect to nuclear, fuel-cycle and safety aspects. This report gives a comparative evaluation of the irradiation behavior of the two fuel types under steady-state and transient conditions, based on the LIFE-III fuel performance code. The analytical results show that properly designed carbide fuel pins will perform as successfully as oxide fuel pins in achieving high burnups on a reliable basis. In addition, for slow and terminated rapid overpower transients, carbide fuels are expected to cause lower levels of FCMI than oxide fuel pins. However, experimental verification of this prediction is required.

- 2960 (WARD-CR--3045-17) CORE RESTRAINT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MAY 31, 1977. Thompson, D.C. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Aug 1977. Contract EY-76-C-02-3045-M. 25p. AT.

Progress related to the development of LMFBR reactor core restraints is summarized in the following areas: three-dimensional development tests, test component friction tests, simulated fuel assemblies, facility modification for full core tests, and TRITON code development. (DG)

- 2961 (WARD-CR--3045-19) CORE RESTRAINT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING AUGUST 31, 1977. Thompson, D.C. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Nov 1977. Contract EY-76-C-02-3045-005. 18p. AT.

Sample Friction Tests to support the 14 Row Test Program were initiated. Evaluation of Phase VI - Addendum C (Low Friction Compaction Test) data was completed. Evaluation of Phase VIII (121 Assembly Straight Core Withdrawal Test) data was initiated. Facility modification for the 14 Row Test was completed. Ninety-five percent of the core assemblies for these tests were assembled and aligned. Work was initiated to extend the next version of TRITON with capabilities of analyzing seven or more assemblies simultaneously. A comparison study of the radial shield assemblies model of TRITON with GHTX model calculations was initiated. Review of the draft for the TRITON code preliminary user's manual was completed.

- 2962 (WARD-CR--3045-20) NATIONAL CORE RESTRAINT DEVELOPMENT PROGRAM: SEVEN ROW TEST RESULTS. Thomson, J.D. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Sep 1977. Contract EY-76-C-02-3045-005. 53p. AT.

The paper describes a test program required to develop the technology necessary to design breeder reactor core restraint systems and support applicable licensing processes. The basis for the qualification of analytical models is described and evaluated relative to the test results. Basic characteristics of the behavior of core assembly arrays are revealed through the implementation of special test methods. An understanding of the mechanical interactions of fuel assemblies in an array is necessary to qualify an analytical model for use as a design tool. "In air" tests were conducted to aid in developing a qualified model. These tests were designed utilizing test

components closely configured to the dimensional, elastic and frictional characteristics of breeder reactor core assemblies in sodium. Comparisons are made between the test results and predictions from an analytical model of the 7 Row test facility. The test facility was modeled using engineering approximations and assumptions characteristic of core restraint design practices. The report, the first of two reports, provides test results from the Westinghouse Core Restraint Test Facility (WCRTF). The report provides information and evaluation on the seven row core array for the following subjects: (1) generalized compaction characteristics, (2) uncertainties related to prediction of core array behavior, (3) comparisons between empirical data and analytical predictions, and (4) statistical modeling basis for core restraint analysis twist effects on core compactions. The results of the evaluation are discussed in terms of additional effort required to fully characterize the effects of mechanical interactions.

2963 (WARD-CR--3045-21) CORE RESTRAINT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING NOVEMBER 30, 1977. Mangus, J.D. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1977. Contract EY-76-C-02-3045-005. 25p. AT.

All core assemblies for the 14 row tests have been aligned and placed on the core. The 14 row characterization tests have been initiated. Documentation of the insertion/withdrawal tests and the statistical modeling work is in progress. This activity must be completed prior to writing the 7 row final report. The Level I-A and I-C sections of the National Core Restraint Code Comparison/Verification Study have been completed using ANSYS 3-D. The TRITON 30° sector model is now operational. The TRITON code user's manual has been reviewed.

2964 (WARD-CR--3045-22) CORE RESTRAINT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING FEBRUARY 28, 1978. Mangus, J.D. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1978. Contract EY-76-C-02-3045-005. 21p. AT.

The LMFBR requires a core restraint system to provide both axial and radial support for the reactor assemblies. The primary requirements of the core restraint system are: (1) fuel, control, blanket, and removable shield assemblies must be located in predictable and reproducible positions; (2) refueling should be accomplished without damage to the assemblies; and (3) operating loads must not be excessive. Variables affecting the core restraint system are primarily the radiation-induced materials effects of swelling and creep which cause bowing deformations of the core assemblies during life and load pad friction. The objective of this development program is to evaluate potential restraint concepts both analytically and empirically, and develop, verify, and qualify analytical tools to be used to evaluate core restraint systems. A test facility, incorporating simulated core assemblies, has been prepared. This facility has the capability of modeling axial and radial core support restraint systems of various BR designs. Results from testing in this facility are used in the development of three-dimensional analytical models for predicting detailed performance of core restraint systems.

2965 (WARD-CR--3045-23) CORE RESTRAINT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR

PERIOD ENDING MAY 31, 1978. Mangus, J.D. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1978. Contract EY-76-C-02-3045-005. 28p. AT.

The LMFBR requires a core restraint system to provide both axial and radial support for the reactor assemblies. The primary requirements of the core restraint system are: (1) fuel, control, blanket, and removable shield assemblies must be located in predictable and reproducible positions; (2) refueling should be accomplished without damage to the assemblies; and (3) operating loads must not be excessive. Variables affecting the core restraint system are primarily the radiation-induced materials effects of swelling and creep which cause bowing deformations of the core assemblies during life and load pad friction. The objective of this development program is to evaluate potential restraint concepts both analytically and empirically, and develop, verify, and qualify analytical tools to be used to evaluate core restraint systems. A test facility, incorporating simulated core assemblies, has been prepared. This facility has the capability of modeling axial and radial core support restraint systems of various BR designs. Results from testing in this facility are used in the development of three-dimensional analytical models for predicting detailed performance of core restraint systems.

2966 (WARD-CR--3045-24) CORE RESTRAINT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING AUGUST 31, 1978. Mangus, J.D. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jun 1979. Contract EY-76-C-02-3045-005. 159p. AT.

Progress is summarized in the following areas: core mechanics and configuration performance; seven-row test evaluations; fourteen row testing; withdrawal/refueling test plan; FFTF load pad dimensional analysis; assessment of proposed Revision 4 creep and Revision 6(7) swelling equations; benchmark problem set for core restraint evaluation of irradiation creep and swelling equations; EBR-2 reflector assembly bowing assessment; swelling-irradiation creep opposition test; FFTF benchmark experiment technical feasibility assessment; TRITON development and verification; ANSYS and CRAMP friction effects evaluations; and national core mechanics and configuration performance plan.

2967 (WARD-CR--94000-1) CORE RESTRAINT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING NOVEMBER 30, 1978. Sirota, B.B. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1978. Contract EY-76-C-02-3045-005. 62p. AT.

Information is presented concerning the Core Restraint Test Facility activities; core restraint structured analysis; and core restraint quality assurance.

2968 (WARD-CR--94000-2) CORE RESTRAINT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING FEBRUARY 28, 1979. Sirota, B.B. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract EY-76-C-02-3045-005. 58p. AT.

Processing of compaction data results from nominal and low friction tests on a straight 7 row core was continued. The 14 row straight core compaction (nested) and dislodged core tests were successfully completed. A numerical comparison of irradiation induced free bow shapes at three points in time indicate ANSYS deflections to be 4%, 17%, and 28% greater than the respective NUBOW-3D predictions; possible

reasons for the differences are being probed. Efforts were continued to provide an analytical evaluation of mechanical effects parameters and assess their significance by comparisons with the mechanical effects resulting from 7 row testing in the Core Restraint Test Facility (CRTF). Effort was started on a benchmark problem set to be used for core restraint evaluation of irradiation creep and swelling equations. Study results of the FFTF benchmark experiment indicate that the test goals are feasible in the multi-duct fuel test assembly (MFTA).

- 2969 (WARD-CR--94000-3) CORE RESTRAINT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MAY 31, 1979. Sirota, B.B. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1979. Contract AM02-76CH94000. 77p. AT.

This quarterly progress report covers the period from March 1, 1979 to May 31, 1979. Processing of compaction data results from nominal and low friction tests on a straight 7 row core was continued. The 14 row straight core testing was focused on specialized compaction tests and assembly characterization measurements during this period. A comparison of irradiation induced free-bow shapes indicated that the NUBOW-3D solution is swelling dominated relative to the ANSYS solution. The first application of a benchmark problem set to be used for core restraint evaluation of irradiation creep and swelling equations was studied for a variety of cases.

- 2970 (WARD-D--0147) INTERNAL EXTERNAL CLADDING DEGRADATION. Travis, M.L. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Feb 1977. Contract EY-76-C-02-3045-020. 66p. AT.

The purpose of the report is to provide information on Internal/External Cladding Degradation. Cladding degradation is used to denote both material loss and degradation of properties due to the high temperature sodium environment and the presence of fuel at high burnups. Various cladding degradation mechanisms that have been identified include: (a) loss of material and surface depletion (chromium and nickel) due to sodium corrosion, (b) degradation of material strength due to loss of carbon and nitrogen, (c) loss of material due to fretting and wear, and (d) internal cladding attack by redistributed fission products.

- 2971 (WARD-D--0178) CLOSURE HEAD CAPABILITY FOR THIRD LEVEL STRUCTURAL MARGIN LOADING. Todd, M.A.; Prevenslik, T.V.; Elder, G.G. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jun 1977. Contract EY-76-C-15-1200. 195p. AT.

The results of the most recent and detailed analysis of response of the Clinch River Breeder Reactor Plant Closure Head System to Third Level Structural Margin Loading are presented and assessed for Closure Head structural adequacy. Results obtained from scale model experiments are summarized and correlated with analytical predictions. Analytical and experimental results both indicate that the Closure Head is capable of withstanding 661 MJ Third Level Structural Margin loads without structural failures.

- 2972 (WARD-HT--3045-27) HIGH TEMPERATURE STRUCTURAL DESIGN TECHNOLOGY: VALIDATION. QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING MAY 31, 1977. Bishop, E.C. (Westinghouse

Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jul 1977. Contract EY-76-C-02-3045. 120p. AT.

Progress is summarized in the testing and analysis of types 304 and 316 stainless steel plates and tubular specimens for the purpose of documenting design criteria for FFTF and LMFBR core structural components. Included are data on core design methods, basic specimen testing, tubular specimen testing, structure testing, FFTF/INX nozzle testing, structural analysis, and crack propagation testing. (DG)

- 2973 (WARD-HT--3045-30) HIGH TEMPERATURE STRUCTURAL DESIGN TECHNOLOGY: VALIDATION. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING AUGUST 31, 1977. Bishop, E.C. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1977. Contract EY-76-C-02-3045-006. 112p. AT.

Progress is summarized in the testing and analysis of types 304 and 316 stainless steel plates and tubular specimens for the purpose of documenting design criteria for FFTF and LMFBR core structural compounds included are data on basic specimen testing, tubular specimen testing, structure testing, FFTF/INX nozzle testing, and structural analysis. (DG)

- 2974 (WARD-HT--3045-31) HIGH TEMPERATURE STRUCTURAL DESIGN TECHNOLOGY: VALIDATION. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING NOVEMBER 30, 1977. Bishop, E.C. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jan 1978. Contract EY-76-C-02-3045-006. 95p. AT.

Progress is summarized in the testing and analysis of types 304 and 316 stainless steel plates and tubular specimens for the purpose of documenting design criteria for FFTF and LMFBR core structural components. Included are data on basic specimen testing, tubular specimen testing, structure testing, FFTF/INX inlet nozzle testing, and analysis methods and procedures.

- 2975 (WARD-HT--3045-34) HIGH TEMPERATURE STRUCTURAL DESIGN TECHNOLOGY: VALIDATION. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING FEBRUARY 28, 1978. Bishop, E.C. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Mar 1979. Contract EY-76-C-02-3045-006. 125p. AT.

Progress is summarized in the testing and analysis of types 304 and 316 stainless steel plates and tubular specimens for the purpose of documenting design criteria for FFTF and LMFBR core structural components. Included are data on basic specimen testing, tubular specimen testing, structure testing, FFTF/INX inlet nozzle testing, and analysis methods and procedures.

- 2976 (WARD-HT--3045-35) RESULTS FROM A BUCKLING TEST OF A 16-INCH (406MM) DIAMETER PIPING ELBOW. Peters, F.E. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jul 1978. Contract EY-76-C-02-3045-006. 228p. AT.

Portions of document are illegible. The report presents a description of a room temperature load test of a 16-inch (406 mm) diameter of Type 304 stainless steel seamless elbow under an in-plane closing bending moment. Strains, forces, displacements and deformations acquired during loading of the elbow test section are presented together with a description of the test, test section and measurement systems. Incremental increases in moment load produced a closing of the elbow

with attendant ovalization of the elbow. Buckling of the elbow was observed at 1.28 x 10⁶ in-lb (144.7 kNm) with no wrinkling of the elbow surface. Peak strains were noted at the elbow crown locations of the central plane (45° section) of the elbow as a result of the elbow ovalization. The report is limited to presentation of test results for comparison with analytical predictions.

- 2977 (WARD-HT--94000-1) STRUCTURAL DESIGN VERIFICATION TESTING TECHNIQUES AT WESTINGHOUSE ADVANCED REACTORS DIVISION. Elliott, D.N.; Horner, R.E.; Peters, F.E. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Apr 1979. Contract EY-76-C-02-3045-006. 54p. AT.

Westinghouse Advanced Reactors Division is performing a program with Department of Energy funding support to validate structural analysis methods and design criteria for piping system components operating in the thermal creep range. The testing techniques discussed in this report relate to full scale tests of AISI 304 stainless steel piping elbows and nozzles. The nominal diameter of the test sections is 16 inch (406 mm), and the tests are performed at temperatures up to 1100°F (593°C). Basic features of the test equipment and test section design are described; however, the emphasis is on instrumentation techniques for measurement of strain and displacement. Instrumentation techniques are discussed for high temperature measurements on the elbow bending tests to determine: strain distribution using capacitance and resistance gages; ovalization at critical cross sections; linear displacement in three directions; rotational displacement and temperature distribution. Strain measurement techniques for nozzle thermal transient tests are discussed, and the selected method is described in detail. Conclusions drawn from this experience with full scale high temperature structural tests are presented in the form of recommendations for related applications.

- 2978 (WARD-HT--94000-2) PLASTIC IN-PLANE BENDING AND BUCKLING OF AN ELBOW: COMPARISON OF EXPERIMENTAL AND SIMPLIFIED ANALYSIS RESULTS. Sobel, L.H.; Newman, S.Z. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Aug 1979. Contract AM02-76CH94000. 200p. AT.

This work shows that analytical predictions obtained from a large displacement MARC pipe-bend element 17 analysis are in reasonably good agreement with experimental results obtained in a companion room temperature test of a 16" OD Type 304 stainless steel elbow, which was loaded by an in-plane closing bending moment in a step-wise fashion until buckling occurred. Estimates of the buckling load obtained from the Southwell Plot Method and a number of presently used arbitrary methods are compared with the experimental buckling load. It is found that the Southwell Method provides an accurate estimate of the buckling load provided that it is based on the nonlinear component of the deformation. The three arbitrary methods considered herein all have the shortcomings of being dependent on the linear elastic region of the moment-deformation curves, whereas the plastic buckling load is not strongly dependent on this region, if at all, and of being devoid of a sound theoretical basis.

- 2979 (WARD-I--10134-2) POST TEST EXAMINATION OF DOUBLE-WALL TUBE FOR INTERFACE WEAR AND WATER SIDE CORROSION EFFECTS IN A LIQUID METAL HEATED STEAM GENERATOR. Padden,

T.R.; Schmidt, D.K. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Dec 1977. Contract EY-76-C-03-0962. 162p. AT.

Post test evaluation of a double-wall steam generator tube of a liquid metal heated 2 1/4 Cr-1 Mo, single tube, J-modular evaporator was performed following a multi-phase test program to determine water side corrosion effects, tube-to-tube interface effects and weld performance. Exposure of the steam generator tubing to 8494 hours of "worst case" water chemistry conditions of 30 ppb sodium under evaporator tube wall "dry out" conditions at 240,000 Btu/hr-ft² heat flux produced no deleterious effects. A conditional maximum corrosion rate of 0.90 mils/year was calculated at the dry-out zone on the basis of corrosion oxide thickness measurements. Tube-to-tube interface wear did occur with a maximum wear depth of 7.9 mils. Tube-to-tubesheet welds and dissimilar metal welds were adequate with no signs of deterioration resulting from the test.

- 2980 (WARD-NA--3045-42) SODIUM TECHNOLOGY PROGRAM: FRICTION, WEAR, AND SELF-WELDING. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JANUARY 31, 1977. Ray, W.E. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Apr 1977. Contract EY-76-C-02-3045. vp. AT.

Testing activities in support of the CRBRP Steam Generator tube/tube spacer plate interaction and Primary Control Rod System (PCRS) material combinations are described. Two tests involving actual 2 1/4 Cr-1 Mo electro-slag remelt (ESR) steam generator tubing rubbing against 2 1/4 Cr-1 Mo and Inconel 718 spacer plate materials were completed in 446°C (835°F) sodium. The tests involved a multiple interface test configuration to permit variations in contact load to be investigated during one test. The final two tests in the steam generator test series have been initiated. Results from tests completed during the reporting period continued to show similar tube wear resulting from interaction with 2 1/4 Cr-1 Mo and Ingot Iron spacer plates. Inconel 718 spacer plate material continued to demonstrate excellent wear behavior when interacting with 2 1/4 Cr-1 Mo tubing at both 446 and 343°C. Initial results appear to indicate little differences in tube wear depth when 2 1/4 Cr-1 Mo spacer plate material interacts with vacuum arc remelt (VAR) or electroslag remelt (ESR) tubing. However, ESR tubing does incur slightly more wear than VAR tubing when interacting with Inconel 718 spacer plate material. Relationships of tube wear depths to contact stress and sodium temperature were also demonstrated.

- 2981 (WARD-NA--3045-48) FRICTION AND WEAR TESTS IN SUPPORT OF LMFBR STEAM GENERATOR TUBE/TUBE SPACER PLATE INTERACTIONS. Galioto, T.A.; Wilson, W.L. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Apr 1979. Contract EY-76-C-02-3045-003. 95p. AT.

The friction, wear, and self-welding behavior of 2-1/4Cr-1Mo steam generator tubing interacting with tube spacer plates has been investigated under prototypic LMFBR steam generator service conditions. Dynamic friction and self-welding characteristics were investigated by a simple pin-on-plate type test in flowing sodium at temperatures to 538°C (1000°F). Subsequent testing involved tube/tube spacer plate model tests utilizing actual steam generator tubing and tube spacer plate materials and configurations. Conditions of contact load, stroke, velocity, rubbing

distance, sodium temperature, and intermittent dwell periods were investigated. In addition, the behavior of potential alternate spacer plate materials Ingot Iron (Armco Electromagnetic Iron) and Inconel 718 as well as Vacuum Arc Remelt and Electroslag Remelt tubing were evaluated.

- 2982 (WARD-NA--3045-49) SODIUM TECHNOLOGY PROGRAM: FRICTION, WEAR, AND SELF-WELDING. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JULY 31, 1977. Ray, W.E. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Dec 1977. Contract EY-76-C-02-3045-001. 57p. AT.

The objectives of the friction and wear test program are: To ensure that no component in the FFTF or CRBRP will fail to perform its function by evaluating the friction and wear behavior of reference and alternate material combinations in interface configurations simulating reactor components under prototypic service conditions and environments, and to provide reactor designers with the quantitative data and experience factors necessary to successfully design components or component interfaces which will experience sliding or static contact under load.

- 2983 (WARD-NA--3045-52) SELF-WELDING EVALUATION OF TRIBALLOY 400, 700, AND 800 IN FLOWING SODIUM. Chang, J.Y.; Schrock, S.L. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Apr 1979. Contract EY-76-C-02-3045-001. 38p. AT.

The objective of this study is to provide experimental information on the self-welding characteristics of materials used in sodium cooled fast breeder reactor components. Materials in some components, such as pumps, valves, and core support structures, will be in static contact, under load, for extended periods while immersed in high-temperature sodium. The self-welding characteristics were determined by measuring the forces required to separate the test specimens in a tensile mode after simulated service exposures to time, temperature, and load in a prototypic sodium environment. Two types of separation tests were performed. In the first type, separation forces were measured at the exposure temperatures in a sodium test chamber (regular test), and the second type was measured at the room temperature in an air atmosphere after removal of the samples from the sodium chamber (screening test).

- 2984 (WARD-NA--3045-53) MICROSTRUCTURAL AND SURFACE CHARACTERIZATION OF COMMERCIAL CANDIDATE LMFBR FUEL CLADDING AND DUCT ALLOYS AFTER EXPOSURE TO FLOWING SODIUM AT 700°C. Bagnall, C.; Witkowski, R.E. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Sep 1978. Contract EY-76-C-02-3045-002. 53p. AT.

A detailed description of the microstructural development of each material at 700°C is given so that the influence of the sodium exposure below the surface exposed to liquid metal may be evaluated. The solute-hardened alloys and the precipitation-hardened alloys displayed distinct and separate types of corrosion. Additional characteristic differences between the two basic groups demonstrate how the concentration of the element molybdenum influences the different surface morphologies that are developed as a result of the sodium exposure.

- 2985 (WARD-NA--3045-54) BEHAVIOR OF AN IRON

MANGANESE ALLOY AS A MONITOR FOR CARBON ACTIVITY IN LIQUID SODIUM. Bagnall, C.; Shiels, S.A. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jun 1978. Contract EY-76-C-02-3045-002. 27p. AT.

Owing to the shortcomings of a number of alloys previously selected for use as carbon activity monitoring materials in sodium systems, a need arose to investigate more fully the behavior of the remaining most promising candidate. Accordingly, DOE funding was provided to study an Fe-12Mn alloy, already identified as the reference material in the RDT Standard "Methods for the Analysis of Sodium and Cover Gas," (RDT-F-3-40T). Some concern had been raised as to the equilibration time and to the degree of response this alloy would have in both high and low carbon activity sodium environments. It was also important to determine how much manganese was lost during foil equilibration since this element has a direct influence on the carbon concentration established at a particular activity level. Four 200 hour tests were performed in an isothermal sodium test loop at 1300°F (704°C) under both carburizing and decarburizing conditions. The results showed that equilibrium was sufficiently established after 24 hours, although the ability of the Fe-Mn foil to carburize to significant levels was not demonstrated. Manganese depletion over a 200 hour period was significant. The surface concentration was lowered to 3-4 percent and the depth of depletion was about 0.8 mils. Several recommendations were made regarding the initial concentration of both manganese and carbon in the standard material, and to a limitation of the sodium exposure period. The need for additional evaluation was indicated.

- 2986 (WARD-NA--940002) PHYSICAL CHARACTERISTICS OF CORROSION PRODUCT DEPOSITS. Miller, R.L. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Sep 1979. Contract AM02-76CH94000. 55p. AT.

Test objectives were to measure the flowrate required to remove deposits from a deposit bearing surface at 700°F and to characterize these sheared deposits. The test technique involved filtering the sodium inventory of a loop, MTL-4 to establish a loop blank, i.e., the quantity of particulates entrained in the sodium. Next, a deposit bearing specimen and filter were installed in MTL-4. Thereafter, the flowrate past the deposit surface was increased in sequential steps with a new filter installed at each test velocity. Comparison of the deposit weights, obtained from the loop blank and test velocity filters, yielded a possible shear velocity for deposit removal. Also the metallic composition and morphology of the filtered debris were determined.

- 2987 (WARD-NC--3045-2) LOOP HEAT CAPACITY MODELS AND THEIR EFFECTS FOR DEMO NATURAL CIRCULATION TRANSIENT ANALYSIS. Planchon, H.P.; Laster, W.R. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Sep 1978. Contract EY-76-C-02-3045-034. 185p. AT.

As part of the DOE Applied Technology Breeder Reactor Natural Circulation Verification Program, improved computer models for the piping, IHX plena, and pump heat transport have been developed and/or adapted for incorporation in DEMO. The improved models include the fluid to surrounding structure heat transfer. Comparisons of the model's calculations with closed form solutions were made for verification. An assessment of the

effect of the fluid to structure heat transfer on the calculations of natural circulation and normal trip transients for the CRBRP configuration, for example, indicate the following: (1) The sodium to structure heat transfer is important when calculating natural circulation thermal heads and flows. The maximum core hot spot temperature immediately following the transition to natural circulation was 200°F higher with loop to sodium heat transfer. Thereafter the heat transfer tends to smooth thermal head and flow variations and reduce peak temperatures. (2) The sodium to structure heat transfer significantly mitigates thermal transients introduced at the reactor, IHX or steam generator. For example, the rate of decrease of temperature at the superheater inlet is reduced by a factor of $1/2$ by the loop heat transfer following a trip to pony motor flow. The mitigation is larger for lower flows and longer pipe runs.

- 2988 (WARD-NC--94000-1) COMPARISONS OF DEMO CALCULATIONS TO FFTF PUMP AND FLOW COASTDOWN MEASUREMENTS. Planchon, H.P.; Calvo, R.; Schurko, R.; Lin, S.S. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Aug 1979. Contract EY-76-C-02-3045-014. 46p. AT.

Main coolant pump speeds and reactor plant flows calculated with DEMO (a whole plant nuclear-thermal-hydraulic transient simulation) are compared to the speed and flow coastdowns, following a pump trip, that were measured in the Fast Flux Test Facility. These comparisons test DEMO's validity for calculating the Reactor Plant coolant flows during the transition from forced circulation to natural circulation. The calculated and measured coastdowns compared closely. The good comparison indicates that DEMO is a valid model for the flow coastdown - that is (1) that the DEMO pump and hydraulic models include the physical elements necessary for calculating the Reactor Plant flow coastdown, and (2) that the DEMO pump and hydraulic models are functioning numerically as intended.

- 2989 (WARD-OX--3045-20) OXIDE FUEL ELEMENT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING SEPTEMBER 30, 1975. Boltax, A.; Saling, J.H. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jan 1976. Contract EY-76-C-02-3045-M. 91p. AT.

Topics reported on include: coordination and evaluation of mixed-oxide irradiation tests; post-irradiation examination of fast reactor experiments; grid-type fuel assembly design; fabrication, development, and testing of LMFBR fuel assemblies; evaluation and analysis of fuel and cladding irradiation behavior; improvement of analytical methods and lifetime codes; and transient code development. (GHT)

- 2990 (WARD-OX--3045-23) OXIDE FUEL ELEMENT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1975. Boltax, A.; Saling, J.H. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Apr 1976. Contract EY-76-C-02-3045-M. 151p. AT.

Progress is summarized in the following LMFBR fuel studies: coordination of mixed oxide irradiation tests; post-irradiation examination of fast reactor experiments; grid-type fuel assembly design; fabrication, development and testing of fuel assemblies; evaluation and analysis of fuel and cladding irradiation behavior; steady-state analytical methods; and transient code development. (DG)

- 2991 (WARD-OX--3045-24) OXIDE FUEL ELEMENT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1976. Boltax, A.; Saling, J.H. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Aug 1976. Contract EY-76-C-02-3045-007;EY-76-C-02-3045-008;EY-76-C-02-3045-009. 110p. AT.

Information is presented concerning the coordination of mixed oxide irradiation tests; post-irradiation examination of fast reactor experiments; grid-type fuel assembly design; fabrication, development, and testing of LMFBR fuel assemblies; evaluation and analysis of fuel and cladding irradiation behavior; steady-state analytical methods; and transient code development.

- 2992 (WARD-OX--3045-26) OXIDE FUEL ELEMENT DEVELOPMENT QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JUNE 30, 1976. Boltax, A.; Saling, J. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1976. Contract EY-76-C-02-3045-007;EY-76-C-02-3045-008;EY-76-C-02-3045-009. 109p. AT.

Progress in LMFBR fuel development and testing is summarized in the following areas: coordination of mixed oxide irradiation tests; post-irradiation examination of fast reactor experiments; grid-type fuel assembly design; fabrication, development, and testing of LMFBR fuel assemblies; evaluation and analysis of fuel and cladding irradiation behavior; steady-state analytical methods; and transient code development. (DG)

- 2993 (WARD-OX--3045-27) OXIDE FUEL ELEMENT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING SEPTEMBER 30, 1976. Boltax, A.; Saling, J. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Nov 1976. Contract EY-76-C-02-3045-007;EY-76-C-02-3045-008;EY-76-C-02-3045-009. 110p. AT.

Information is presented concerning the coordination of mixed oxide irradiation tests; post-irradiation examination of fast reactor experiment; 217 rod grid-type fuel assembly mechanical design; fabrication, development, and testing of LMFBR fuel assemblies; evaluation and analysis of fuel and cladding irradiation behavior; steady-state analytical methods; and transient code development.

- 2994 (WARD-OX--3045-29) OXIDE FUEL ELEMENT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING DECEMBER 31, 1976. Boltax, A.; Saling, J. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jun 1979. Contract EY-76-C-02-3045-007;EY-76-C-02-3045-008;EY-76-C-02-3045-009. 82p. AT.

Progress is summarized for the following tasks: coordination of mixed-oxide irradiation tests, post-irradiation examination of fast reactor experiments, evaluation and analysis of fuel and cladding irradiation behavior, radial blanket irradiation tests in EBR-2, grid type fuel assembly mechanical design, grid fabrication and tests, and fuel pin design criteria and code development.

- 2995 (WARD-OX--3045-31) OXIDE FUEL ELEMENT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1977. Boltax, A.; Cooper, M.H. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jun 1979. Contract EY-76-C-02-3045-007;EY-76-

C-02-3045-008;EY-76-C-02-3045-009. 125p. AT.

Progress is summarized for the following tasks: coordination of mixed-oxide irradiation tests, post-irradiation examination of fast reactor experiments, evaluation and analysis of fuel and cladding irradiation behavior, radial blanket irradiation tests in EBR-2, grid type fuel assembly mechanical design, grid fabrication and tests, and fuel pin design criteria and code development.

2996 (WARD-OX--3045-32) OXIDE FUEL ELEMENT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JUNE 30, 1977. Boltax, A.; Cooper, M.H. (Westinghouse Electric Corp., Pittsburgh, PA (USA). Advanced Reactors Div.). Jun 1979. Contract EY-76-C-02-3045-007;EY-76-C-3045-008;EY-76-C-02-3045-009. 100p. AT.

Progress is reported in the following studies: coordination of mixed-oxide irradiation tests, post-irradiation examination of fast reactor experiments, evaluation and analysis of fuel and cladding irradiation behavior, radial blanket irradiation tests in EBR-2, grid-type fuel assembly mechanical design, grid fabrication and tests, and fuel pin design criteria and code developments. (JWR)

2997 (WARD-OX--3045-34) OXIDE FUEL ELEMENT DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING SEPTEMBER 30, 1977. Boltax, A.; Cooper, M.H. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jun 1979. Contract EY-76-C-02-3045-007;EY-76-C-02-3045-008;EY-76-C-02-3045-009. 113p. AT.

Studies on LMFBF grid-type fuel assembly mechanical design include mixed-oxide irradiation tests; post-irradiation examination of fast reactor experiments; evaluation and analysis of fuel and cladding irradiation behavior; radial blanket irradiation tests in EBR-2; grid type fuel assembly mechanical design; fabrication, development, and testing of breeder reactor fuel assemblies; and fuel pin design criteria and code development.

2998 (WARD-OX--3045-35) APPLICATION OF LIFE TRANSIENT CODES TO EXPERIMENT AND CORE DESIGN. Roth, T.S.; Sundquist, B.E.; Biancheria, A. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Feb 1978. Contract EY-76-C-02-3045-009. 26p. AT.

The LIFE series of codes model cylindrical fuel pin behavior for various types of fast breeder reactor fuels. Results from the transient codes which analyze off-normal, non-core disruptive events up to cladding breach are discussed. The results reported are from initial applications of the code to experiment and core design. The core design application involved analysis of mixed oxide fuel pin performance during design basis upset and emergency transients. The upset events considered produce moderate cladding damage. The emergency event is more severe, consuming about 60 percent of the cladding lifetime. Sensitivities of damage to variations in fabricated gap size, cladding swelling, and thermal hydraulic environment were examined. A mixed carbide 50 cents/sec transient experiment to be performed in TREAT was studied focusing on the peak power necessary to produce incipient failure. Sensitivities of cladding damage and strain to variations in initial fuel cladding mechanical interaction, thermal hydraulic environment, and fuel hot pressing were examined.

2999 (WARD-OX--3045-36) HYDRAULIC AND VIBRATION CHARACTERIZATION RESULTS OF PROTOTYPIC WATER FLOW TESTS OF A 217-ROD GRID SPACED RIVER ASSEMBLY. Akey, J.G.; Schwallie, A.L. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Nov 1977. Contract EY-76-C-02-3045-008. 420p. AT.

The detailed results are presented of the data reduction for the flow induced vibration and mechanical response tests. The validity of these results are evaluated, and the application of these results and their conclusions to FFTF operation is discussed. These objectives are compatible with the original test objectives of defining the dynamic characteristics of the fuel assembly in air and stagnant water, of measuring the induced vibration of the model fuel rods, and of predicting the vibration characteristics of the driver fuel assembly based upon analysis of the model flow test data.

3000 (WARD-OX--3045-39) BREEDER REACTOR GRID-TYPE SUBASSEMBLY DESIGN AND EVALUATION. QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING DECEMBER 31, 1977. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Apr 1978. Contract EY-76-C-02-3045-008. 20p. AT.

Studies on LMFBF grid-type fuel assembly mechanical design include ductless core design and the fabrication, development, and testing of assemblies. Objectives, prior work, and current progress in these studies are summarized.

3001 (WARD-OX--3045-40) BREEDER REACTOR GRID-TYPE SUBASSEMBLY DESIGN AND EVALUATION. QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING MARCH 31, 1978. Cooper, M.H. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1978. Contract EY-76-C-02-3045-008. 36p. AT.

In the grid development program the effort is directed toward development of a fuel pin spacer system that will allow comparison with the wire-wrap system. The grid-type subassemblies will be tested in the first core of the FTR, where a one-to-one comparison of the two types of spacer systems can be made. The results of the analytical and experimental studies on experiments conducted in EBR-II will be fully utilized in this program.

3002 (WARD-OX--3045-45) EFFECTS OF TEMPERATURE CHANGES ON 20% COLD WORKED AISI 316 STAINLESS STEEL SWELLING. Foster, J.P.; Boltax, A. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Feb 1979. Contract EY-76-C-02-3045-009. 23p. AT.

Large swelling values associated with a gradual temperature reduction during irradiation have been observed for pressurized tubes irradiated at two initial irradiation temperatures. A comparison of the pressurized tube data with WSA-4 fuel pin cladding swelling data shows that the same behavior is observed in fuel pins subject to temperature changes during irradiation. The fuel pin data cover a large temperature interval and this information was used to determine temperature and fluence dependencies of the observed enhanced swelling effect. The results show that the enhanced swelling with temperature reductions depends on the initial irradiation temperature with the maximum value occurring at about 500°C. The data indicate that the effect saturates with fluence.

- 3003 (WARD-PB--3045-1) FTF HYDROSTATIC PUMP BEARING PHASE-1 TEST REPORT (FINAL REPORT). (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). [nd]. Contract EY-76-C-02-3045-014. vp. AT.

In-sodium tests were conducted to provide pertinent data related to various performance characteristics of a prototypic FTF sodium lubricated hydrostatic pump bearing. The characteristics evaluated included: (1) bearing load/deflection performance, (2) wear and galling resistance during bearing rubbing at low speeds and (3) the cleanability of the bearing assembly after exposure to a sodium environment. The test program provided the required load/deflection and cleanability data to the pump design organization as required; however, there is an undetermined degree of uncertainty associated with the wear characteristics data due to problems encountered with the test rig.

- 3004 (WARD-RB--3045-19) RADIAL BLANKET DESIGN AND DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MAY 31, 1977. Bishop, E.C. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jul 1977. Contract EY-76-C-02-3045. 78p. AT.

Progress in LMFBR radial blanket studies is summarized in the following areas: heat transfer testing, irradiation testing, flow orificing tests, and rod bundle compaction tests. (DG)

- 3005 (WARD-RB--3045-21) RADIAL BLANKET DESIGN AND DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING AUGUST 31, 1977. Bishop, E.C. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Sep 1977. Contract EY-76-C-02-3045-M. 87p. AT.

Summaries of prior work and current progress on LMFBR radial blanket design and development are presented for the following: heat transfer testing, irradiation testing, flow orificing test, and rod bundle compaction tests.

- 3006 (WARD-RB--3045-22) RADIAL BLANKET DESIGN AND DEVELOPMENT QUARTERLY PROGRESS REPORT FOR PERIOD ENDING NOVEMBER 30, 1977. Bishop, E.C. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jan 1978. Contract EY-76-C-02-3045-004. 74p. AT.

Summaries of prior work and current progress are presented for the following task areas: (1) radial blanket assembly heat transfer testing, (2) radial blanket assembly irradiation, (3) radial blanket flow orificing, and (4) radial blanket rod bundle compaction tests. (DG)

- 3007 (WARD-RB--3045-24) RADIAL BLANKET DESIGN AND DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING FEBRUARY 28, 1978. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jul 1978. Contract EY-76-C-02-3045-004. 92p. AT.

Progress is summarized in the following areas: radial blanket assembly heat transfer testing, radial blanket assembly irradiation, radial blanket flow orificing, and radial blanket rod bundle compaction tests. (DG)

- 3008 (WARD-RB--3045-29) RADIAL BLANKET DESIGN AND DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MAY 31, 1978.

- Bishop, E.C.; Yingling, J.E. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jun 1979. Contract EY-76-C-02-3045-004. 85p. AT.

Information is presented concerning radial blanket assembly heat transfer testing; radial blanket assembly irradiation; radial blanket flow orificing; and radial blanket rod bundle compaction tests.

- 3009 (WARD-RB--3045-30) RADIAL BLANKET DESIGN AND DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING AUGUST 31, 1978. Bishop, E.C.; Yingling, J.E. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jun 1979. Contract EY-76-C-02-3045-004. 79p. AT.

Information is presented concerning radial blanket assembly heat transfer testing; radial blanket assembly irradiation; radial blanket flow orificing; and radial blanket rod bundle compaction tests.

- 3010 (WARD-RB--3045-31) RADIAL BLANKET DESIGN AND DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING NOVEMBER 30, 1978. Bishop, E.C. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jun 1979. Contract EY-76-C-02-3045-004. 80p. AT.

Information is presented concerning the radial blanket assembly heat transfer test; radial blanket assembly irradiation; radial blanket flow orificing; and radial blanket rod bundle compaction tests.

- 3011 (WARD-RB--94000-1) RADIAL BLANKET DESIGN AND DEVELOPMENT. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING FEBRUARY 28, 1979. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Aug 1979. Contract EY-76-C-02-3045-011. 74p. AT.

Summaries of prior work and current progress are presented for the following task areas: (1) radial blanket heat transfer testing; and (2) blanket assembly irradiation.

- 3012 (WARD-SD--3045-7) EFFECTS OF GEOMETRIC NONUNIFORMITIES ON THE STRESSES IN PIPING ELBOWS. Thomas, K. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jan 1978. Contract EY-76-C-02-3045-032. 76p. AT.

A comparative evaluation is presented of the effects of thickness variation, initial ovality, intrados wrinkling and longitudinal weld shrinkage on thin-walled elbow stresses under pressure and in-plane moment loadings. This comparison will identify the nonuniformities of greatest significance and will also provide a basis for evaluation of the Code method of determining stress indices for the case of initial ovality. The analysis method and analytical models developed using the STAGS program are described. The results for each of the nonuniformities investigated are presented.

- 3013 (WARD-SD--3045-8) ANALYSIS OF TESTS ON AXIALLY COMPRESSED CYLINDERS. Newman, S.Z.; Sobel, L.H. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Nov 1977. Contract EY-76-C-02-3045-032. 34p. AT.

Four axial compression room temperature tests on 3.5 inch (8.9 cm) outer diameter by 0.095 inch (0.24 cm) thick AISI 304 stainless steel cylinders are documented. The cylinders are loaded to buckling and beyond, and their

deformation behavior is described. Two uniaxial tensile tests are performed on coupon specimens to determine material properties, and the results are given. The coupon specimens along with the four tested cylinders were cut from longer tubes, and they were all tested in the "as-received" condition.

- 3014 (WARD-SD--3045-9) PLASTIC BUCKLING OF AXIALLY COMPRESSED CYLINDERS. Dhalla, A.K.; Sobel, L.H. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Nov 1977. Contract EY-76-C-02-3045-032. 89p. AT.

Analytical and experimental results are presented for the problem of plastic buckling of axially compressed cylindrical shells. The cylinders are made from strain hardening materials and their geometries (radius-to-thickness ratio $r/t = 10$ to 50) are in the range of interest encountered in the design of LMFBR reactor vessels, heat exchangers, and piping systems. The primary objective of this work is to determine which one (if any) of a number of simple closed form formulas for the buckling load is reasonably accurate, as all of the formulas are very convenient for design and scoping analysis purposes. The determination is based on a comparison of buckling loads predicted by the various formulas (simplified analysis) with those obtained from tests and from the more rigorous nonlinear buckling method of analysis (detailed analysis).

- 3015 (WARD-SD--3045-10) HIGH TEMPERATURE STRUCTURAL DESIGN TECHNOLOGY: APPLICATION. SEMIANNUAL PROGRESS REPORT FOR THE PERIOD ENDING SEPTEMBER 30, 1977. Griffin, D.S. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Mar 1979. Contract EY-76-C-02-3045-032. 216p. AT.

Progress in the development of high-temperature structural design methods for LMFBR components is reported in the following areas: design criteria and standards; simplified inelastic analysis methods; computer program validation and utilization; and piping integrity.

- 3016 (WARD-SD--3045-11) PLASTIC COLLAPSE OF A PIPING ELBOW: CONVERGENCE OF A FINITE ELEMENT SHELL ANALYSIS. Dhalla, A.K. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jul 1978. Contract EY-76-C-02-3045-032. 206p. AT.

Deformation characteristics of a representative thin walled elbow were evaluated using a fully compatible doubly-curved isoparametric thin shell element of the MARC program. Geometric nonlinear shell analyses indicate that the solution converges monotonically from the stiff side, with element refinement and incremental load step refinement. Mesh refinement and load step sizes are established to compute the collapse load of an elbow structure. The overall deformation behavior of the elbow is classified into four categories, (a) linear elastic, (b) nearly linear elastic-plastic, (c) elastic-plastic with slight geometric nonlinearity, and (d) elastic-plastic substantially nonlinear. This classification may simplify selection of an analysis method in the design of a piping network.

- 3017 (WARD-SD--3045-12) USER'S GUIDE TO MAT3D: A THREE-DIMENSIONAL FINITE ELEMENT MESH GENERATOR PROGRAM FOR EIGHT-NODE ISOPARAMETRIC ELEMENTS. Kent, M.P.; Newman, S.Z.; Griffin, D.S. (Westinghouse Electric Corp., Madison,

PA (USA). Advanced Reactors Div.). Oct 1978. Contract EY-76-C-02-3045-032. 82p. AT.

The MAT3D computer program, described herein, is a mesh generator for the APACHE (thermal) and MATUS (structural) three-dimensional finite element computer programs. APACHE and MATUS use an eight-node, tri-linear, isoparametric element. The data required to describe a mesh composed of an assemblage of such elements consist of the three coordinates associated with each node and the eight nodes associated with each element. MAT3D automatically generates this data from a lesser amount of information provided by the user. A visual means of checking the correctness of the generated geometry is provided by MAT3D in the form of graphical output plots.

- 3018 (WARD-SD--3045-13) USER'S GUIDE TO APACHE: A THREE-DIMENSIONAL FINITE ELEMENT PROGRAM FOR STEADY-STATE OR TRANSIENT HEAT CONDUCTION ANALYSIS. Griffin, D.S. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1978. Contract EY-76-C-02-3045-032. 56p. AT.

The APACHE computer program was designed to solve three-dimensional nonlinear steady-state or transient heat conduction problems, using the finite element method. Although APACHE is an independent program, it was written primarily as a thermal companion to the MATUS structural analysis program in order to provide a complete three-dimensional thermo-mechanical analysis capability. The MAT3D program (originally called MESH3D) can be used to generate finite element meshes for use in APACHE and MATUS. The capabilities and application of the APACHE program are described.

- 3019 (WARD-SD--3045-16) DESIGN GUIDELINES FOR ELBOWS WITH GEOMETRIC NONUNIFORMITIES. Thomas, K. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jan 1979. Contract EY-76-C-02-3045-032. 94p. AT.

The report presents methods and design guidelines for evaluating the effects of certain elbow nonuniformities under internal pressure loading. These nonuniformities are thickness variation, initial ovality, intrados wrinkling, and longitudinal weld shrinkage. The ultimate purpose of such an investigation is to enable designers to determine what level of nonuniformity can be tolerated in any given design and by this means form a rational basis for specification of a fabrication method. Two main aspects of this task emerge from the investigation, namely, the development of methods to determine the effects of the nonuniformities and the embodiment of these results in the design criteria. Thickness variation and wrinkling are covered by means of limits on the amount of nonuniformity while initial ovality is allowed for by determining the stress effects. The results of the present work could form the basis of additions to the criteria. These would give the designer the option of using these results to determine stress effects of the nonuniformity or using the existing criteria. Of course, in the case of longitudinal weld shrinkage, no rules exist currently.

- 3020 (WARD-SD--94000-1) CREEP TENSILE INSTABILITY. O'Donnell, W.J.; Porowski, J.S. (O'Donnell and Associates, Inc., Pittsburgh, PA (USA)). Jul 1979. Contract EY-76-C-02-3045-032. 46p. AT.

Creep tensile instability solutions are obtained for most cases encountered in design

applications. These include: (1) membrane load stresses of arbitrary biaxiality, (2) pressurized spherical vessels, and (3) pressurized cylindrical vessels and piping with combined axial loads. Solutions are obtained for both the effective (Mises) strains and the maximum tensile strains at instability. The results show that neither Triaxiality Factors nor factors based on plastic tensile instability solutions can be used to convert creep rupture ductility data to safe allowable design values for components operating in the creep regime. The present solutions can be used to obtain allowable strains for design applications using available uniaxial creep rupture ductility data.

- 3021 (WARD-SD--94000-2) MATLIB: A COMPUTER LIBRARY OF MATERIAL PROPERTIES VERIFICATION MANUAL. Konish, H.J.; Snow, A.L. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jul 1979. Contract EY-76-C-02-3045-042. 239p. AT.

The difficulties inherent to the analysis and evaluation of structural behavior in the elevated temperature regime may be significantly reduced by the use of the MATLIB library of computer programs and subroutines for characterizing material behavior at elevated temperatures. This library provides the analyst with a large selection of descriptions of material behavior in the elevated temperature regime and a variety of design limits with which predicted structural response may be compared to assess structural adequacy. These items are provided in a form that facilitates ready access and widespread general use. One of the principal virtues of MATLIB is that each item has been subjected to extensive verification prior to release for general use. The term verification, as used herein, refers to the correct execution of specified calculational procedures. The verification of MATLIB items is described in this report.

- 3022 (WARD-SR--3045-6) EFFECT OF REPAIR TIMES ON RELATIVE SHRS RELIABILITY. Bertucio, R.C. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Nov 1978. Contract EY-76-C-02-3045-029. 67p. AT.

The impact of maintainability on the relative reliability of various Shutdown Heat Removal System Configurations on loop type LMFBR's is studied. The study found the Reactor Auxiliary Cooling (RAC) configurations maintain their reliability advantage over the Primary Reactor Auxiliary Cooling (PRACS) and Intermediate Reactor Auxiliary Cooling (IRACS); although this advantage is reduced by factors of three and four respectively. Sensitivity studies indicate the RACS will maintain its reliability advantage for all repair times of practical interest. The study also found that dependence on two loops to remove decay heat (rather than assuming each loop has 100% capacity) increases the unreliability by a factor of 500 to 7000 for the RACS and by a factor of 100 to 500 for the PRACS and IRACS. The RACS still maintains a reliability advantage, however.

- 3023 (WARD-SR--3045-7) LARGE LMFBR SHUTDOWN HEAT REMOVAL SYSTEM RELIABILITY TRADE-OFF STUDY. Bertucio, R.C. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Feb 1977. Contract EY-76-C-02-3045-029. 102p. AT.

Various shutdown heat removal systems were evaluated to make a preliminary estimate of which configurations provide the most reliable

method of post-shutdown heat removal. Six design concepts were considered. Each concept was evaluated with and without a safety-related auxiliary condenser and feedwater system in the main heat transport system rejecting heat through the steam generators. For each design configuration, a Failure Mode and Effects Analysis was completed at the component level. This was used to make a qualitative judgment of reliability and to provide input to a Fault Tree Analysis. Predictions of numerical annual unreliability were completed. Each of the six designs were ranked relative to each other.

- 3024 (WARD-SR--3045-8) COMPARISON OF EXPERIMENT WITH ANALYSIS OF A WATER-FILLED PLASTICALLY DEFORMING PIPE USING THE COMPUTER CODE PTA-1. Mak, M. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1978. Contract EY-76-C-02-3045-029. 45p. AT.

In a liquid metal cooled fast breeder reactor, pressure pulses resulting from hypothetical core disruptive accidents (HCDA) or sodium/water reactions may plastically deform the thin-walled piping typically used in the sodium systems. The plastic deformation has a significant effect on pressure-transient propagation since it limits the peak pressure of the fluid transmitted out of a pipe to the pressure approximately causing the pipe to yield if the pipe is sufficiently long. PTA-1 is a computer code developed by the Argonne National Laboratory for computing the transmission of pressure pulses through the system and the effect of plastic deformation of the piping on pressure-transient propagation. In order to provide experimental data to validate computer codes developed for the safety evaluations of reactors, Stanford Research Institute has completed a series of experiments of simple piping systems (Reference 2). One of these experiments, the flexible straight pipe test FP-SP-191, has been adopted here to verify the PTA-1 computed responses. Details of this test are described.

- 3025 (WARD-SR--3045-9) ASSESSMENT OF DEBRIS TRANSPORT METHODS IN HORIZONTAL LMFBR PIPING FOR PAHR CONDITIONS. Brown, R.G. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Apr 1979. Contract EY-76-C-02-3045-029. 59p. AT.

In assessing the long-term coolability of debris following a postulated core disruptive accident, the distribution of the fragmented fuel and steel must first be determined. One area which has received relatively little attention is the potential for debris transport into the primary piping and the distribution of settled debris in the piping system. A review of existing methods for analyzing particle transport in horizontal pipes has been performed to assess the adequacy of these techniques for LMFBR post-accident heat removal (PAHR) evaluations. It was concluded that the existing methods for analyzing particle settling are inappropriate for the PAHR conditions of high density ratio, low concentration mixtures flowing in large pipes. Therefore, a modified correlation has been developed to predict the critical deposition velocity for solid-liquid flow in horizontal pipes for conditions of interest in PAHR evaluations.

- 3026 (WARD-SR--94000-1) NUCLEAR SAFETY AND RELIABILITY ENGINEERING. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING MARCH 31, 1979. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Apr 1979.

Contract EY-76-C-02-3045-031. 63p. AT.

Progress is summarized in the following LMFBF safety-related areas: (1) safe shutdown reliability program, (2) safety analysis in support of low risk design capability, (3) shutdown system reliability testing, (4) inherent shutdown system development, (5) safety analysis in support of limited core damage, (6) safety analysis in support of core debris accommodation, (7) low risk containment design criteria, (8) control of radiological consequences, and (9) radial blanket assembly flow blockage test.

3027 (WARD-SR--94000-2) NUCLEAR SAFETY AND RELIABILITY ENGINEERING. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JUNE 30, 1979.

(Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Jul 1979. Contract EY-76-C-02-3045-031. 59p. AT.

Progress is summarized in the following LMFBF safety-related areas: (1) safe shutdown reliability, (2) safety analysis in support of low risk design capability, (3) shutdown system reliability testing, (4) inherent shutdown system development, (5) safety analysis in support of limited core damage, (6) safety analysis in support of core debris accommodation, (7) low risk containment design criteria, and (8) control of radiological consequences.

3028 (WARD-XS--3045-18) PHYSICS EVALUATIONS

AND APPLICATIONS. Arakali, V.; Davis, W.J.; Paik, N.C. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). May 1977. Contract EY-76-C-02-3045. 15p. AT.

Progress is summarized in the following areas: implementation of the VENTURE code on the Westinghouse NES CDC-7600 computer, Doppler effects on power transients initiated by a step reactivity insertion in a typical LMFBF, and analysis of blanket composition to minimize blanket power variation. (DG)

3029 (WARD-XS--3045-19) PHYSICS EVALUATIONS AND APPLICATIONS QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JULY 31, 1977.

Yarbrough, M.; Arakali, V.; Davis, W.J. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Aug 1977. Contract EY-76-C-02-3045-012. 30p. AT.

Information is presented concerning analysis of the ZPR-6-7 benchmark problems for the RRD code evaluation working group; implementation of the VENTURE code on the Westinghouse NES CDC-7600 computer; calculation of breeding ratio and sodium void worth in a large LMFBF using (U-Th) oxide and (U-Th) carbide fuels; and ERDA meeting on physics requirements to support alternate fuel cycles.

3030 (WARD-XS--3045-20) PHYSICS EVALUATIONS AND APPLICATIONS. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING OCTOBER 31, 1977.

Arakali, V.; Davis, W.J. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Mar 1978. Contract EY-76-C-02-3045-012. 27p. AT.

Progress is reported for the following activities: (1) Advanced System analyses of the heterogeneous core design concept for small and large LMFBFs; (2) Full implementation and utilization of the ERDA supported three dimensional burnup diffusion theory code, VENTURE, at ARD for the design verification required for the CRBRP FSAR and future LMFBF designs; (3) Utilization of the ERDA developed two-dimensional kinetics code, FX-2, for

transient analyses and HCDA energetics studies of advanced LMFBF design concepts; (4) Continued evaluations and applications of the ENDF/B-nuclear data files for general LMFBF analyses; and (5) Continued participation in the Cross Section Evaluation Working Group (CSEWG) and the Computer Code Coordination and Method Testing committees.

3031 (WARD-XS--3045-21) PHYSICS EVALUATIONS AND APPLICATIONS. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JANUARY 31, 1978. Arakali, V.S.; Calamai, G.J.; Yarbrough, M.B. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Apr 1978. Contract EY-76-C-02-3045-012. 30p. Dep. NTIS, PC A04/MF A01.

Progress is reported for the following activities: (1) Advanced System analyses of the heterogeneous core design concept for small and large LMFBFs; (2) Full implementation and utilization of the ERDA supported three dimensional burnup diffusion theory code, VENTURE, at ARD for the design verifications and development of future tools required for LMFBF designs; (3) Utilization of the ERDA developed two-dimensional kinetics code, FX-2, for the transient analyses and HCDA energetics studies of advanced LMFBF design concepts; (4) Continued evaluations and applications of the ENDF/B-nuclear data files for general LMFBF analyses, and (5) Continued participation in the Cross Section Evaluation Working Group (CSEWG), the Computer Code Coordination and Method testing committees.

3032 (WARD-XS--3045-22) PHYSICS EVALUATIONS AND APPLICATIONS. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING APRIL 30, 1978. Arakali, V.S.; Paik, N.C. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Aug 1978. Contract EY-76-C-02-3045-012. 21p. AT.

Progress is summarized in studies of delayed neutron constants for large heterogeneous LMFBF cores and design analysis of an intermediate-size LMFBF. (DOG)

3033 (WARD-XS--3045-23) PHYSICS EVALUATIONS AND APPLICATIONS QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JULY 31, 1978. Paik, N.C.; Rathbun, R.W.; Varner, S.K.; Petras, D.S.; Yarbrough, M.B. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Oct 1978. Contract EY-76-C-02-3045-012. 25p. AT.

Progress is summarized in the following task areas: (1) advanced system analyses of the heterogeneous core design concept for small and large LMFBFs; (2) full implementation and utilization of the DOE supported three-dimensional burnup diffusion theory code, VENTURE, at ARD for the design verifications and development of future tools required for LMFBF designs; (3) utilization of the DOE developed two-dimensional kinetics code, FX-2, for the transient analyses and HCDA energetics studies of advanced LMFBF design concepts; (4) continued evaluations and applications of the ENDF/B-nuclear data files for general LMFBF analyses; and (5) continued participation in the Cross Section Evaluation Working Group (CSEWG), and the Computer Code Coordination and Method testing committees.

3034 (WARD-XS--3045-24) PHYSICS EVALUATIONS AND APPLICATIONS QUARTERLY PROGRESS REPORT FOR PERIOD ENDING OCTOBER 31, 1978. Yarbrough, M.B.; Calamai, G.J. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors

Div.). Mar 1979. Contract EY-76-C-02-3045-012. 39p. AT.

During this quarter, Westinghouse further quantified the disc usage charge components during the VENTURE analysis of the large 84 x 42 x 33 mesh, 9 group test problem. Previous work with the large test problem had shown that approx. 45% of the total job cost was attributable to storing and retrieving information from disc. Westinghouse therefore studied this problem in a three-step analysis program. First, a parametric study was performed to determine the impact of varying the disc file allocation and transfer size parameters, via the CDC-7600 job control language specifications. Second, the data transfers themselves were monitored during the flux solution iterations to identify the key cost components. Finally, the results of the two previous steps were combined to identify areas where modifications to the data transfer strategy may be beneficial to utilization of the code at Westinghouse.

3035 (WARD-XS--3045-25) PHYSICS EVALUATIONS AND APPLICATIONS QUARTERLY PROGRESS REPORT FOR PERIOD ENDING JANUARY 31, 1979. Paik, N.C.; Varner, S.K.; Yarbrough, M.B. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). 1979. Contract EY-76-C-02-3045-012. 21p. AT.

Under this advanced FBR system development task, ARD has initiated a study to determine the neutronic feasibility of a new and improved proliferation and diversion resistant FBR. This unique FBR design concept utilizes U^{235} fuel and incorporates an operating mode which does not require any refueling throughout the lifetime of the plant. These design features significantly enhance its proliferation and diversion resistance characteristics. Furthermore, it does not require an operating plutonium reprocessing plant.

3036 (WARD-XS--94000-1) PHYSICS EVALUATIONS AND APPLICATIONS. QUARTERLY PROGRESS REPORT FOR PERIOD ENDING APRIL 30, 1979. Yarbrough, M.B.; Varner, S.K. (Westinghouse Electric Corp., Pittsburgh, PA (USA). Advanced Reactors Div.). Jun 1979. Contract AM02-76CH94000. 15p. AT.

Current progress in LMFBR physics evaluations is reported in the following areas: three-dimensional perturbation module implementation and testing and a non-refueling ^{235}U -fueled LMFBR.

3037 (WHAN-IR--33) COOLANT MIXING IN A 217-PIN WIRE-WRAPPED FFTF FUEL ASSEMBLY. Reihman, T.C.; Hildner, R.A. (WADCO Corp., Richland, WA (USA)). Sep 1970. Contract EY-76-C-14-2170. 68p. AT.

An experimental study using a direct sampling technique was performed, with water as the test fluid, to determine the interchannel coolant mixing in a 217-pin wire-wrapped FFTF fuel assembly model. A comparative computer study was made to determine individual subchannel exchange rates from the experimental measurements.

3038 (WNET--116) RELIABILITY OF VARIOUS TUBE-TO-TUBESHEET WELD CONFIGURATIONS. Flury, R.L.; Morales, W.J.; York, J.W. (Westinghouse Electric Corp., Tampa, FL (USA). Tampa Div.). Jun 1975. Contract EY-76-C-03-0962. 59p. Dep. NTIS, PC A04/MF A01.

Through the application of FMEA and FTA Techniques the reliabilities of various tube-to-tubesheet weld configurations consisting of

different combinations single or duplex tubes and single or double tubesheets with or without leak detection are investigated for application to LMFBR steam generator design. Based on substantial failure experience accumulated on PWR units, the study confirms the high reliability these welds possess and in particular identifies the tube-to-boss fillet weld as the most efficient, reliable design.

3039 (ZfK--325) DETERMINATION OF DYNAMIC PARAMETERS OF THE REACTOR BOR-60 BY MEANS OF A REACTIVITY MODULATOR. Krause, H.; Otto, H.; Scholz, J.; Ziegenbein, D.; Afanasjew, W.; Britov, W.; Grjasew, W.; Jefimow, W.; Keadse, B. (Zentralinstitut fuer Kernforschung, Rossendorf bei Dresden (German Democratic Republic)). Feb 1977. 20p. (In German). Dep. NTIS (US Sales Only), PC A02/MF A01.

It is reported about first transfer function measurements performed at the reactor BOR-60 in the range 0.05 to 18 cps and at powers up to 45 MW by means of a rotating reactivity modulator. The construction of the modulator is described roughly. Its reliability is treated. By fitting the results of the measurements to a simple model for the heat transfer in the reactor it could be shown, that the dynamics of the reactor for frequencies >0.05 cps may be described by one power coefficient with the adequate time constant.

3040 (ZfK--337) PROBLEMS OF TECHNOLOGY AND CORROSION IN SODIUM COOLANT AND COVER GAS. Kuenstler, K.; Ullmann, H. (eds.). (Zentralinstitut fuer Kernforschung, Rossendorf bei Dresden (German Democratic Republic)). Jul 1977. 293p. (In Russian). (CONF-7703105--1). Dep. NTIS (US Sales Only), PC A09/MF A01.

From Meeting on problems of technology and corrosion in sodium coolant and cover gas; Dresden, German Democratic Republic (28 Mar 1977).

The meeting encloses the following themes: (i) Reactions in the system sodium-steel-cover gas (ii) Corrosion behaviour of structural and cladding materials (iii) Determination of impurities in sodium and cover gas (iv) Technology of sodium and cover gas (v) Testing equipments (vi) Safety problems.

3041 (ZfK--344) SODIUM BOILING EXPERIMENT IN THE REACTOR BOR-60. Afanasjew, V.A.; Mitenkov, F.M.; Fuge, R. (Zentralinstitut fuer Kernforschung, Rossendorf bei Dresden (German Democratic Republic)). Oct 1977. 9p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Sodium boiling was generated in the BOR-60 core by γ -heating of a tungsten rod bundle in a special test assembly. The principal possibility of boiling detection by acoustic and neutron noise was shown. Some results of spectrum and correlation analysis are presented and the necessity for tailored signal evaluation methods in order to increase the reliability of boiling detection is pointed out.

3042 (ZfK--348) PERTURBATION MEASUREMENTS ON THE BOR-60 REACTOR. Kaetzmer, D.; Maletti, R.; Pohls, J.; Ziegenbein, D. (Zentralinstitut fuer Kernforschung, Rossendorf bei Dresden (German Democratic Republic)). Dec 1977. 11p. (In German). Dep. NTIS (US Sales Only), PC A02/MF A01.

It is reported about first perturbation measurements on the reactor BOR-60. The measurements were made on different reactor states, also in the state with sodium boiling

in the central cassette. The results correspond to former results, which were received by reactivity balance methods and animate to further researches. It was found, that the boiling does not influence the reactivity value of control rods and the power coefficient of reactivity.

- 3043 (ZPR-TM--348) EXPERIMENTAL STUDIES OF 350 MW(E) HETEROGENEOUS LMFBR CORES AT ZPPR. Collins, P.J.; McFarlane, H.F.; Beck, C.L.; Lineberry, M.J.; Carpenter, S.G.; Ducat, G.A.; Gasidlo, J.M.; Goin, R.W. (Argonne National Lab., IL (USA)). 5 Sep 1979. Contract W-31-109-ENG-38. 19p. (CONF-790933--2). Dep. NTIS, PC A02/MF A01.

From IAEA international symposium on fast reactor physics; Aix-en-Provence, France (24 Sep 1979).

The annular heterogeneous concept has been adopted for the Clinch River Breeder Reactor. To provide verification of design and safety analysis, fourteen core configurations were studied at ZPPR. The results and analysis of the early assemblies, ZPPR-7A, 7B, and 7C, have been previously reported and only the more significant conclusions are summarized here. Later assemblies, 7D to 7H, were designed to provide data on variations of control rod and internal blanket arrangements. Assemblies 8A to 8E provided data on the replacement of uranium by thorium in various blanket subassemblies.

- 3044 (ZPR-TM--349) EXPERIMENTAL STUDIES OF LARGE CONVENTIONAL LMFBR CORES AT ZPPR. Lineberry, M.J.; Carpenter, S.G.; Beck, C.L.; McFarlane, H.F.; Collins, P.J.; Olsen, D.N.; Grasseschi, G.L.; Kaiser, R.E.; Maddison, D.W. (Argonne National Lab., Idaho Falls, ID (USA)). 5 Sep 1979. Contract W-31-109-ENG-38. 22p. (CONF-790933--4). Dep. NTIS, PC A02/MF A01.

From IAEA international symposium on fast reactor physics; Aix-en-Provence, France (24 Sep 1979).

Results of a critical experiment program for 700 MW(e) LMFBRs are reviewed. Both a clean benchmark assembly and one simulating a power reactor design were constructed. Experiments discussed include reaction rate distributions and ratios, control rod worths and interactions, and sodium coefficients. Analysis of these experiments with ENDF/B Version IV cross sections is reported. The comparison between calculated and measured parameters is not too different in these cores than in the 350 MW(e) assemblies. An exception is that the radial fission rate is calculated up to 3% too high at the edge of the inner core relative to the core center.

- 3045 (PNL-TR--400) REVIEW OF THE DEVELOPMENT OF FAST BREEDER REACTORS IN THE USSR. Kochetkov, L.A.; Bagdasarov, Yu.E. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). 21 Apr 1980. Translation source information not available 12p. Dep. NTIS, PC A02/MF A01.

Information on fast breeder reactors in the USSR is presented concerning the BR-5 (BR-10) reactor; BOR-60 reactor; BFS-2 test rig; BN-350 reactor; BN-600 reactor; and BN-1600 reactor.

- 3046 (PNL-TR--398) STUDY OF THE OPERATIONAL CHARACTERISTICS OF THE CORE AND IRRADIATED FUEL PARAMETERS IN THE BOR-60 REACTOR. Gryazev, V.M.; Afanasiev, V.A.; Gadzhiev, G.I.; Efimov, V.N.; Inchagov, A.V.; Markin, S.A.; Nigmatullin, N.P.; Telkovtsev, A.A. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov,

Dimitrovgrad (USSR)). 14 Apr 1980. Translation of IAEA-SM--225/67. 14p. NTIS, PC A02/MF A01.

In order to determine the operational lifetimes of various fuels in terms of charge, temperature conditions, burn-up and integrated neutron flux, an extensive irradiation program for various types of fuel compositions and structural materials was carried out on the BOR-60 reactor. Ninety-percent enriched uranium dioxide in the form of sintered toroids with 10.2 g/cm³ density was used as the standard fuel. The claddings of 37 fuel elements 6 mm in diameter are made of 0X16H15M3B austenite class stainless steel. In all, 225 such assemblies (approx. 8300 fuel elements) were irradiated, including 194 assemblies with burn-up of over 8% heavy atoms. Maximum burn-up was 13.2% heavy atoms, neutron flux was 9.1×10^{22} neutrons/cm² (E > 0.1 MeV), linear power attained 540 W/cm, and cladding temperature reached 700°C.

- 3047 (PNL-TR--397) ENSURING SAFE OPERATION OF THE BOR-60 EXPERIMENTAL FAST REACTOR AS AN ATOMIC POWER PLANT. Gryazev, V.M.; Antipin, G.K.; Afanasiev, V.A.; Britov, V.N.; Efimov, V.N.; Kondrat'ev, V.I.; Kon'kov, V.F.; Smirnov, A.M. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). 14 Apr 1980. Translation of IAEA-SM--225/68, pp. 177-189. (CONF-780407--10). 23p. NTIS, PC A02/MF A01.

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

On the basis of the BOR-60 facility an analysis is made of characteristic requirements for safe operation of a fast-reactor power plant, from the standpoint of stability against perturbations in the operation of the main components and systems and against deviations from the design values of process parameters. The self-regulation properties of the facility are discussed. Regulation by neutron flux and by sodium outlet temperature are compared from the point of view of increasing the reliability of the control system and minimizing temperature strains in the components during transient processes. Calculated and experimental data are presented on optimization of emergency shut-down and cooling regimes using different combinations of equipment. The means and methods used to verify the state of the reactor vessel, fuel assemblies and in-vessel reactor systems during operation on power and during reloading are described: operating experience with sodium/water steam generators is analyzed, as is the use of leak verification, control and protection for steam generators under accident conditions.

- 3048 (PNL-TR--393) USE OF DEMONSTRATION FAST REACTOR EXPERIENCE IN ASSESSING THE RADIATION SAFETY OF FAST POWER REACTORS IN THE DESIGN STAGE. Bakumenko, O.D.; Kiselev, G.V.; Kulakovskii, M.Ya.; Pomerantsev, G.B.; Samarkin, A.A.; Sergeev, V.A.; Skorikov, N.V.; Tverdovskii, N.D.; Tsurukin, Yu.P. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). 4 Apr 1980. Translation of IAEA-SM--225/65, 1978. (CONF-780407--11). 10p. NTIS, PC A02/MF A01.

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The results are presented of experimental studies on the behavior of fission products and radioactive corrosion products in the primary circuit of the BN-350 reactor during operation

at power levels of up to 650 MW with defective fuel elements in the core. Data are also given on radiation conditions in the reactor building, dose burden to personnel and activity loss from the primary circuit. The experimental data thus obtained are compared with design data.

3049 FISSION GAS RELEASE AND MICROSCOPIC SWELLING IN HIGHLY RATED ADVANCED FUELS.

Coquerelle, M.; Walker, C.T. (Commission of the European Communities, Karlsruhe, Germany). Nucl. Technol.; 48: No. 1, 43-53(Apr 1980).

Mixed carbide, carbonitride, and nitride fuels have been irradiated in DFR and Rapsodie to a maximum burnup of 7.8 at. % at a maximum linear power of 135 kW/m burnup, xenon release from helium-bonded fuels was found to be dependent on the chemical composition of the fuel. Release was greatest from carbide (75%) and least from nitride fuels (35%). At medium burnup, improved gap conductance led to a fall in the fuel centerline temperature and consequently a decrease in gas release. For nitride and carbonitride fuels, over 75% of the retained fission gas was contained in bubbles (<1 µm in diameter) and in the fuel matrix. For all three fuels, xenon release from the outer unrestructured region of the fuel was <15%, whereas release from the central porous region was 50% or more. In the restructured region, gas was released to the plenum by way of interconnected pores. Gas in pores contained proportionally more krypton than the bonded gas, and consequently, it is proposed that atomic diffusion is the principal mechanism of gas transport within the fuel.

3050 PLENUM-2A, A PROGRAM FOR TRANSIENT ANALYSIS OF LIQUID-METAL FAST BREEDER REACTOR OUTLET PLENUMS. Carbajo, J.J.; Howard, P.A. (Argonne National Lab., IL). Nucl. Technol.; 47: No. 2, 244-256(Feb 1980).

A new computer program, PLENUM-2A, has been developed to predict the conditions in the outlet plenum of a liquid-metal fast breeder reactor after a transient scram. This program divides the transient into three main consecutive discrete states. Two additional stages, one occurring before the transient begins and the other occurring as the transient ends, are also considered. The three main stages are described in detail, and calculations and experimental results are compared. The agreement between the calculations and experimental data is very good.

3051 ULTRASONIC INSPECTION OF LIQUID-METAL FAST BREEDER REACTOR STEAM GENERATOR DUPLEX TUBING. Longua, K.J.; Whitham, G.K.; Allen, C.C. (Argonne National Lab., Idaho Falls). Nucl. Technol.; 47: No. 2, 257-267(Feb 1980).

Two ultrasonic inspections of the Experimental Breeder Reactor II steam generator duplex tubing have been completed. Inspections performed on one evaporator in 1976 provided baseline data, and a subsequent inspection in 1978 revealed no change in tube condition. With the completion of the 1978 inspection, all available tubes in one evaporator have been inspected. The steam generator contains duplex tubes fabricated from 2 1/4 Cr-1 Mo ferritic steel. Access to the bore (water) side of the tubes was gained through the steam outlet piping. The inspection included a complete volumetric (100% of the tube material) examination, measurement of wall thickness, and evaluation of the condition of the braze bonding the two walls of the tube together. The test equipment was routinely calibrated against

a standard containing artificial flaws. Artificial flaws as small as 1.6 mm long x 0.25 mm deep were readily detected.

3052 EVAPORATIVE REMOVAL OF SODIUM FROM REACTOR COMPONENTS. Welch, F.H.; Steele, D.P. III. (Rockwell International Energy Systems Group, Canoga Park, CA). Nucl. Technol.; No. 2, 282-307(Feb 1980).

Evaporation of sodium from components using heat and vacuum was investigated as a method of sodium removal from liquid-metal fast breeder reactor components. The method was determined to be capable of efficiently cleaning components at temperatures as low as 260 to 315°C and at pressures of approx. 1×10^{-3} to 1×10^{-4} Pa. The most rapid cleaning was observed with small crevices, where the sodium is expelled by gases released from the sodium. The method was recommended as a viable process that should be investigated using full-size components.

3053 CONTAINMENT RESPONSE TO POSTULATED CORE MELTDOWN ACCIDENTS IN THE FAST FLUX TEST FACILITY. Gasser, R.D.; Pratt, W.T. (Brookhaven National Lab., Upton, NY). Nucl. Technol.; 47: No. 2, 282-307(Feb 1980).

An assessment is made of the containment margin available in the Fast Flux Test Facility to mitigate the consequences of a postulated failure of in-vessel post-accident heat removal following a hypothetical core disruptive accident. The consequences of a number of assumed meltdown configurations (both in-vessel and ex-vessel) are assessed using the CACECO (CAvity, CEll, COntainment) containment analysis computer code together with currently available melt front penetration models. The sensitivity of the accident scenarios to a number of crucial assumptions is established by scoping studies. It is concluded from both the in-vessel and exvessel analyses that sodium vapor combustion is a major source of reactor containment building (RCB) pressurization. The conditions (a combination of sodium-concrete reaction, pool size, and decay heat level) that most rapidly bring the sodium to boiling, together with those that enhance mass transfer of sodium vapor to the RCB, are the ones that most significantly affect the pressure response.

3054 NUCLEAR REACTOR WITH FUEL PIN BRACING GRID. Jolly, R. (to United Kingdom Atomic Energy Authority). US Patent 4,181,571. 1 Jan 1980. Priority date 9 Sep 1977, United Kingdom of Great Britain and Northern Ireland (UK), 6p.

A fuel pin bracing grid for a nuclear fuel sub-assembly comprises a honeycomb array of unit cells formed from discrete strips. The cells are hexagonal, three alternate sides having windows and the remaining sides have linear groups of three embossments to provide guide pads for fuel pins. The openings provide a measure of compliance for the grid to facilitate insertion and withdrawal of the pins. A fuel sub-assembly for a liquid metal cooled fast breeder nuclear reactor has a central fuel section with end extensions, the fuel section comprising a bundle of fuel pins in a hexagonal wrapper the pins being braced by a series of grids according to the invention. Reprocessing of the fuel is facilitated because the pins are withdrawable collectively from the compliant grids and wrapper combination merely by cutting an end extension from the wrapper. 4 claims.

3055 NUCLEAR REACTOR CONTROL. Ingham, R.V. (to United Kingdom Atomic Energy Authority). US Patent 4,181,569. 1 Jan 1980. Priority date 13 Apr 1976, United Kingdom of Great Britain and Northern Ireland (UK), 4p.

A liquid metal cooled fast breeder nuclear reactor has power setback means for use in an emergency. On initiation of a trip-signal a control rod is injected into the core in two stages, firstly, by free fall to effect an immediate power-set back to a safe level and, secondly, by controlled insertion. Total shut-down of the reactor under all emergencies is avoided. 4 claims.

3056 ANALYSIS OF THE DIFFERENCES IN BREEDING RATIO AND FISSILE INVENTORY BETWEEN HETEROGENEOUS AND HOMOGENEOUS LIQUID-METAL FAST BREEDER REACTORS. Tzanos, C.P. (Science Applications, Inc., Oak Brook, IL). Nucl. Technol.; 47: No. 1, 39-50 (Jan 1980).

The differences in fissile inventory and breeding ratio, with respect to the differences in fertile inventory and neutron spectrum, between equivalent heterogeneous and homogeneous configurations were analyzed. To quantify the effect of spectral changes on reaction rate ratios, a calculational scheme based on properly prepared one-group cross-section sets was used.

3057 EFFECTS OF TEMPERATURE CHANGES ON THE SWELLING BEHAVIOR OF 20% COLD-WORKED AISI TYPE 316 STAINLESS STEEL. Foster, J.P.; Boltax, A. (Westinghouse Electric Corp., Madison, PA). Nucl. Technol.; 47: No. 1, 181-188 (Jan 1980).

Large swelling values associated with a gradual temperature reduction during irradiation have been observed for pressurized tubes irradiated at two initial irradiation temperatures. A comparison of the pressurized tube data with WSA-4 fuel pin cladding swelling data shows that the same behavior is observed in fuel pins subject to temperature changes during irradiation. The fuel pin data cover a large temperature interval, and this information was used to determine temperature and fluence dependencies of the observed enhanced swelling effect. The results show that the enhanced swelling with temperature reductions depends on the initial irradiation temperature, with the maximum value occurring at approx. 500°C. The data indicate that the effect saturates with fluence.

3058 IRRADIATION BEHAVIOR OF EXPERIMENTAL MARK-II EXPERIMENTAL BREEDER REACTOR II DRIVER FUEL. Hofman, G.L. (Argonne National Lab., IL). Nucl. Technol.; 47: No. 1, 7-22 (Jan 1980).

Prototypic driver-fuel elements using metallic fuel and stainless-steel cladding, designed to achieve a high burnup, were tested in the Experimental Breeder Reactor II. The irradiation results showed that burnup of up to 10 at.% can be attained without cladding failure and that cladding deformation can be kept to acceptable values if Type 316 stainless steel is used as the cladding material.

3059 EFFECT OF INTRASUBASSEMBLY INCOHERENCIES ON THE FAST TEST REACTOR UNPROTECTED TRANSIENT OVERPOWER ACCIDENT. Yung, S.C.; Wilburn, N.P. (Hanford Engineering Development Lab., Richland, WA). Nucl. Technol.; 47: No. 1, 23-38 (Jan 1980).

Intrasubassembly incoherencies affecting the fuel pin failure pattern within a fast test reactor (FTR) subassembly during an unprotected transient overpower/hypothetical core

disruptive accident have been investigated using the COBRA-III/MELT code. Two dominant intrasubassembly incoherencies in an FTR subassembly were studied, namely, (a) the hydraulic effect, or the variation in pin-power-to-effective-coolant ratio between pins in the inner region and those in the outer region of the subassembly, and (b) the power skew, or variation in pinwise power density for pins throughout the subassembly.

3060 GRAVITATIONAL COLLISION EFFICIENCY OF POST HYPOTHETICAL CORE DISRUPTIVE ACCIDENT LIQUID-METAL FAST BREEDER REACTOR AEROSOLS: SPHERICAL PARTICLES. Pertner, G.A.; Loyalka, S.K. (Univ. of Missouri, Columbia). Nucl. Technol.; 47: No. 1, 70-90 (Jan 1980).

Sensitivity analyses have shown that the gravitational collision efficiency influences post hypothetical core disruptive accident aerosol behavior in liquid-metal fast breeder reactor (LMFBR) containment in important ways. Research was directed toward improving expressions for this quantity. Following the work of atmospheric sciences, dynamical equations for two particle motions were developed. A computer program GCEFF was constructed, options for using a variety of drag forces were provided, and the dynamical equations were solved by using Gear's method. Results were compared with the previous work of atmospheric sciences, and explicit results for several cases of interest in the LMFBR studies were provided.

3061 ROLE OF FISSION GAS AND FUEL MELTING IN FUEL RESPONSE DURING SIMULATED HYPOTHETICAL LOSS-OF-FLOW TRANSIENTS. Bandyopadhyay, G.; Buzzell, J.A. (Argonne National Lab., IL). Nucl. Technol.; 47: No. 1, 91-109 (Jan 1980).

Direct electrical heating (DEH) experiments have been performed to study fuel and fission gas behavior during transients with thermal conditions similar to those predicted for flow-coastdown and sodium voiding phases of a reference reactor hypothetical loss-of-flow accident case. Macroscopic fuel response, such as gross fuel swelling and fuel dispersal in DEH fuel pellet stacks, was monitored during the transients. It was noted that in the presence of a mild restraint (e.g., due to quartz "cladding"), fuel melting always occurred prior to any detectable gross fuel motion in the stack. The fuel response at failure was strongly dependent on the thermal history of the simulated flow-coastdown phase and the heating rates during the subsequent phase of the transient experiments. In the presence of a mild restraint, the thermal history before fuel melting occurred in the stack strongly influenced the fuel behavior.

3062 SELF LOCKING DRIVE SYSTEM FOR ROTATING PLUG OF A NUCLEAR REACTOR. Brubaker, J.E. US Patent 4,178,208. 11 Dec 1979. Filed date 23 Feb 1978. 8p.

PAT-APPL-880,679.

A self locking drive system for rotating the plugs on the head of a nuclear reactor which is able to restrain plug motion if a seismic event would occur during reactor refueling is described. A servomotor is engaged via a gear train and a bull gear to the plug. Connected to the gear train is a feedback control system which allows the motor to rotate the plug to predetermined locations for refueling of the reactor. The gear train contains a self locking double enveloping worm gear set. The worm gear set is utilized for its self locking nature to prevent unwanted rotation of the plugs as the result of an earthquake. The double enveloping

type is used because its unique contour spreads the load across several teeth providing added strength and allowing the use of a conventional size worm.

3063 FLOW MEASUREMENT IN MAIN SODIUM PIPES.

Hans, R.; Knaak, J.; Weiss, H.J. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). At. Energy Rev.; 17: No. 4, 803-890 (Dec 1979).

The problems of sodium flow measurement in the main circuits of LMFBRs in the 1000MW power range are discussed and existing techniques reviewed with respect to their application to pipes of large nominal bore. The application conditions are summarized in the context of both pool and loop-type reactors. Electromagnetic, ultrasonic and transit-time measuring techniques are covered. In addition to the theoretical bases of the measurement - in particular for the reactor designer who is not familiar with the theoretical background - application aspects are discussed including accuracy, response speed, installation and maintenance. The merits and demerits of the different methods for particular reactor designs are outlined. The possibilities of redundancy and diversity are also discussed. Finally, the present state of development of these methods - as far as published - is summarized.

3064 FLOW BEHAVIOR OF VOLUME-HEATED BOILING POOLS: IMPLICATIONS WITH RESPECT TO TRANSITION PHASE ACCIDENT CONDITIONS.

Ginsberg, T. (Brookhaven National Lab., Upton, NY); Jones, G.C. Jr.; Chen, J.C. Nucl. Technol.; 46: No. 3, 391-398 (Dec 1979).

Observations of two-phase flow fields in single-component volume-heated boiling pools were made. Photographic observations, together with pool-average void fraction measurements, indicate that the churn-turbulent flow regime is stable for superficial vapor velocities up to nearly five times the Kutateladze dispersal limit. Within this range of conditions, a churn-turbulent drift flux model provides a reasonable prediction of the pool-average void fraction data. An extrapolation of the data to transition phase accident conditions suggests that intense boilup could occur where the pool-average void fraction would be >0.6 for steel vaporization rates equivalent to power levels $>1\%$ of nominal liquid-metal fast breeder reactor power density. The extended stability of bubbly flow to unusually large vapor fluxes and void fractions, observed in some experiments, is a major unresolved issue.

3065 THERMAL-HYDRAULIC ANALYSIS OF FUEL BLOCKAGES FORMED DURING TRANSIENT OVERPOWER ACCIDENTS IN LIQUID-METAL FAST BREEDER REACTORS.

Dhir, V.K.; Kastenber, W.E.; Varela, D.W. (Univ. of California, Los Angeles). Nucl. Technol.; 46: No. 3, 447-452 (Dec 1979).

Generally, the blockages that may form in the reactor core when molten fuel is ejected into the coolant channels during a transient overpower accident can be divided into two categories. The first type of blockage forms when semi-molten fuel particles hit a wire wrap and freeze on the wire wrap, whereas the second type of blockage forms when solid or semi-solid particles get stuck in the coolant channels. The thermal-hydraulic analysis of these proposed blockages indicates that under certain specific conditions, the blockages can lead to melting of the wire wrap and cladding and also to in-channel voiding. The second type of

blockage, if formed as a plug, may remelt before being quenched by subcooled or saturated sodium.

3066 COMPARATIVE STUDY OF HETEROGENEOUS AND HOMOGENEOUS LIQUID-METAL FAST BREEDER REACTOR CORES IN SOME ACCIDENT CONDITIONS.

Renard, A. (Belgonucleaire Rue du Champ de Mars, Brussels, Belgium); Evrard, G.; Wehmann, U. Nucl. Technol.; 46: No. 3, 453-464 (Dec 1979).
The behavior of a heterogeneous and a homogeneous design of a liquid-metal fast breeder reactor core with the same power and similar dimensions at beginning-of-equilibrium cycle is compared for several accident conditions (loss-of-flow and transient overpower) using the same failure criteria and model assumptions. The influence of the design is analyzed, and qualitative trends are deduced from the point of view of safety.

3067 CRITICAL EXPERIMENTAL STUDY OF INTEGRAL PHYSICS PARAMETERS IN SIMULATED LIQUID-METAL FAST BREEDER REACTOR MELTDOWN CORES.

Bhattacharyya, S.K.; Wade, D.C.; Bucher, R.G.; Smith, D.M.; McKnight, R.D.; LeSage, L.C. (Argonne National Lab., IL). Nucl. Technol.; 46: No. 3, 517-524 (Dec 1979).

Integral physics parameters of several representative, idealized meltdown liquid-metal fast breeder reactor (LMFBR) configurations were measured in mockup critical assemblies on the Zero Power Reactor 9 at Argonne National Laboratory. The experiments were designed to provide data for the validation of analytical methods used in the neutronics part of LMFBR accident analyses. Large core distortions were introduced in these experiments (involving 18.5% core volume), and the reactivity worths of configuration changes were determined. The neutronics parameters measured in the various configurations showed large changes on core distortion. Both diffusion theory and transport theory methods were shown to mispredict the experimental configuration eigenvalues. In addition, diffusion theory methods were shown to result in a nonconservative misprediction of the experimental configuration change worths.

3068 LOW SODIUM VOID CORES.

Barthold, W.P.; Beitel, J.C.; Lam, P.S.K.; Drechwa, Y.; Su, S.F.; Turski, R.B. (Argonne National Lab., IL). Nucl. Technol.; 46: No. 3, 525-528 (Dec 1979).
To avoid high energy releases in unprotected loss-of-flow accidents, cores can be designed such that the removal of sodium would add only a small amount of reactivity or even a negative reactivity to the reactor. Reduction in sodium void reactivity can be achieved by changing either the geometry or composition of the core. Pancake, modular, and heterogeneous core configurations were investigated. Heterogeneous cores showed sodium void reactivities in the 2-dollar range with only small penalties in doubling time when compared with the equivalent homogeneous cores. Liquid-metal fast breeder reactors using U-Th fuel in the form of metal, oxide, or carbide show negative sodium void reactivities but doubling times above 30 yr.

3069 LMFBR STEAM GENERATORS: BEHAVIOR OF HEAT EXCHANGE TUBES FACED WITH A THROUGH CRACK RESULTING IN CONTACT BETWEEN SODIUM AND WATER.

Quinet, J.L.; Lannou, L. (Electricite de France, Chatou). Nucl. Technol.; 46: No. 3, 582-590 (Dec 1979).
A survey on the behavior of cracked tubes under operating conditions of an industrial steam generator was conducted. A comparison was

first made of the development kinetics of the cracks according to the type of material: Incoloy 800, 2-1/4 Cr-1 Mo, 9 Cr-2 Mo, and to the initial leak. Then, two successive stages are evidenced in the self-development process of a water leak into sodium: (1) an incubation stage, during which the material is subject to intense corrosion in the crack area; and (2) a self-erosion stage, marked by the appearance of a leak rate that grows slowly at first, then suddenly, when the tube wall near the crack gives way under pressure.

- 3070 FUEL PINS AND CORE RESPONSE UNDER LIQUID-METAL FAST BREEDER REACTOR TRANSIENT OVERPOWER ACCIDENT CONDITIONS. Wilburn, N.P.; Smith, D.E.; Baars, R.E.; Atcheson, D.B.; Spencer, B.W. (Hanford Engineering Development Lab., Richland, WA). Nucl. Technol.; 46: No. 3, 591-599(Dec 1979).

Since the earlier liquid-metal fast breeder reactor transient overpower assessments were done (1975), new experimental data and modeling improvements have occurred that indicate later failures and more molten fuel squirted into the channel with a higher propensity for plugging. An initial sweepout still occurs, and an analysis shows that even if coherent instead of the expected stochastic failures occur, the blockages are partial, the reactor is strongly shut down, and a coolable geometry exists. Hence, the overall consequences would be benign.

- 3071 ANALYSIS OF TRANSIENT OVERPOWER ACCIDENTS IN THE FAST TEST REACTOR USING THE LOS ALAMOS FAILURE MODEL. Mast, P.K.; Scott, J.H. (Los Alamos Scientific Lab., NM). Nucl. Technol.; 46: No. 3, 600-605(Dec 1979).
A new fuel pin failure model, the Los Alamos Failure Model (LAFM), based on a linear life fraction rule failure criterion, has been developed to provide a reliable and inexpensive prediction of the time and axial location of liquid-metal fast breeder reactor fuel pin failure in a hypothetical transient overpower (TOP) accident. Code testing analyses for a number of TOP Transient Reactor Test Facility tests have resulted in excellent agreement between calculated and observed pin failure time and location. Because of the nature of the failure criterion used, the code has also been used to investigate the extent of cladding damage incurred in terminated as well as unterminated TOP transients in the Fast Test Reactor. The results of these analyses show that 3 dollar/s and 50 and 5 cent/s transients terminated by the secondary trip point (25% overpower) result in negligible calculated cladding damage.

- 3072 CONTROLLING CESIUM IN THE COOLANT OF THE EXPERIMENTAL BREEDER REACTOR II. Olson, W.H. (Argonne National Lab., Idaho Falls, IA); Ruther, W.E. Nucl. Technol.; 46: No. 2, 318-322(Dec 1979).

Carbon has been shown to be an effective getter for cesium in liquid sodium. A cesium trap containing reticulated vitreous carbon and operating at 190°C has been used at the Experimental Breeder Reactor II to reduce fission product cesium activity in the primary sodium coolant from a maximum of 15 kBq/g to approx. 870 Bq/g.

- 3073 MODES OF FAILURE OF FAST REACTOR FUEL PINS IRRADIATED UNDER LOSS-OF-COOLANT-FLOW CONDITIONS. Kwast, H. (Netherlands Energy Research Foundation ECN, Petten). Nucl. Technol.; 46: No. 2, 234-240(Dec 1979).

Capsule irradiations have been performed on single fast reactor fuel pins in a sodium environment under simulated loss-of-coolant-flow conditions. The main objectives were to determine the thresholds, modes, and mechanisms of fuel pin failures. The parameters were canning temperature and internal pin pressure. The loss-of-coolant-flow condition was simulated by adjusting midwall canning temperatures of approx. 850 and approx. 1000°C. The results indicated that the creep rupture is the predominant failure mechanism at canning temperatures of 1000°C and gas pressures of above 40 bars. The failure mechanism of fuel pins tested at approx. 850°C and gas pressures lower than 60 bars is probably cladding strain due to differential expansion of fuel and canning.

- 3074 INVESTIGATION OF LOCAL COOLING DISTURBANCES IN AN IN-PILE SODIUM LOOP IN THE BR2. Kramer, W. (Kernforschungszentrum Karlsruhe, Germany); Schleisiek, K.; Schmidt, L.; Vanmassenhove, G.; Verwimp, A. Nucl. Technol.; 46: No. 2, 281-288(Dec 1979).

The fuel element behavior under local off-normal cooling conditions and the possible pin-to-pin failure propagation are of special interest in the safety analysis of liquid-metal fast breeder reactors. In a program called Mol 7C, Kernforschungszentrum Karlsruhe and the Centre d'Etude de l'Energie Nucleaire/ Studiecentrum voor Kernenergie at Mol are performing related experiments in a sodium loop in the Br2 reactor. The test section contains a 37-pin bundle of UO₂ fuel with an artificial local blockage involving the 6 inner pins and parts of the pins of the second row (30 rods are active fuel pins, and 7 are stainless-steel dummy pins). After a pre-irradiation of some days, the transient test phase is initiated by interrupting the cooling in the local blockage region. The first and second of three planned experiments demonstrate that the cooling of the whole fuel bundle was not jeopardized, although several pins have failed. The performance of the third experiment had to be postponed for the end of 1979 due to the exchange of the beryllium matrix of the BR2 reactor during 1979.

- 3075 CONSIDERATIONS ON INCOHERENCY OF BOILING AND VOIDING IN LIQUID-METAL FAST BREEDER REACTOR SUBASSEMBLIES DURING A LOSS-OF-FLOW ACCIDENT. Nijsing, R.; Eifler, W. (Commission of the European Communities, Ispra, Italy). Nucl. Technol.; 46: No. 2, 289-299(Dec 1979).

The present version, THARC-8, of the transient thermohydraulic subassembly code, THARC, under development at the Joint Research Centre at Ispra, has been applied to predict the transient thermohydraulic behavior of liquid-metal fast breeder reactor subassemblies subjected to loss-of-flow conditions. Information is given on the present status of development of THARC. Computational results are presented for 217-rod subassemblies, both for grid and wire spacers. It is shown that radial power gradients and overcooling in the region adjacent to the wrapper wall cause boiling inception in the subassembly to be incoherent with, as a consequence, a more gradual boiling and voiding process than expected on the basis of a one-dimensional approach. It is demonstrated that heat capacity effects associated with the fuel rods and the wrapper wall are of considerable importance. Aspects associated with code validation in out-of-pile bundle experiments are also briefly discussed.

- 3076 ATTENUATION OF AIRBORNE DEBRIS FROM LIQUID-METAL FAST BREEDER REACTOR ACCIDENTS. Morewitz, H.A. (Rockwell International, Canoga Park, CA); Johnson, R.P.; Nelson, C.T.; Vaughan, E.U.; Guderjahn, C.A.; Hilliard, R.K.; McCormack, J.D.; Postma, A.K. Nucl. Technol.; 46: No. 2, 332-339(Dec 1979).

Experimental and theoretical studies have been performed to characterize the behavior of airborne particulates (aerosols) expected to be produced by hypothetical core disruptive accidents (HCDAs) in liquid-metal fast breeder reactors (LMFBRs). These aerosol studies include work on aerosol transport in a 20-m-high, 850-m³ closed vessel at moderate concentration; aerosol transport in a small vessel under conditions of high concentration (approx. 1 kg/m³), high turbulence, and high temperature (approx. 2000°C); and aerosol transport through various leak paths. These studies have shown that little, if any, airborne debris from LMFBR HCDAs would reach the atmosphere exterior to an intact reactor containment building.

- 3077 IN-REACTOR EXPERIMENTS ON THE COOLING OF FAST REACTOR DEBRIS. Rivard, J.B. (Sandia Labs., Albuquerque, NM). Nucl. Technol.; 46: No. 2, 344-349(Dec 1979).

First-of-a-kind fission-heated experiments utilizing uranium oxide particles in liquid sodium have been performed to assess the nature of the passive heat transfer between fast reactor fuel debris and overlying coolant. The experiments were designed to simulate the situation following a core disruptive accident in which molten core material is quenched, fragmented, and is dispersed as beds of decay-heated particulate within the reactor vessel. In two of the experiments, threshold dryout of the fuel particulate was produced. During several runs, dryout was maintained for long periods (approx. 1 h) with only modest temperature increases, demonstrating that while bed dryout may be a necessary condition for remelting of the fuel, it is not always a sufficient condition.

- 3078 MULTICHANNEL MODEL FOR FUEL-COOLANT INTERACTION IN A LIQUID-METAL FAST BREEDER REACTOR SUBASSEMBLY. Mehr, K. (Joint European Research Center, Ispra, Italy); Kottowski, H.M.; Goldammer, H. Nucl. Technol.; 46: No. 2, 362-368(Dec 1979).

A multichannel model and computer code for fuel-coolant interaction in a liquid-metal fast breeder reactor subassembly have been developed. The model takes into account the geometry of the subassembly and assumes a partial meltdown. Results obtained with the code show the influence of local initial perturbations. Furthermore, it is revealed that the common vapor space between the coolant and the molten fuel is the predominant coupling between the channels.

- 3079 IMPROVED TREATMENT OF THE NEUTRON STREAMING THROUGH CONTROL ROD FOLLOWERS IN A SODIUM-COOLED FAST REACTOR. Yoshida, T. (Inst. für Neutronenphysik und Reaktortechnik, Karlsruhe, Germany). Nucl. Sci. Eng.; 72: No. 3, 361-365(Dec 1979).

A method is presented for determining by numerical integration a diffusion coefficient, $D_{\text{sub } z}$, applicable to the axial direction of a control rod follower region in a sodium-cooled fast reactor. By comparing criticality values from transport and diffusion theory for a simplified reactor model, it is shown that this $D_{\text{sub } z}$ applies well to the followers from the viewpoint of the calculated

criticality factor, $k_{\text{sub eff}}$. By use of the same model, an intercomparison is also made between the present and other definitions of $D_{\text{sub } z}$. By using the present $D_{\text{sub } z}$, a conclusion is that the currently used D ($= 1/3 \Sigma$) for control rod followers leads to an underestimation of $k_{\text{sub eff}}$ by approx. 0.3% for a typical 300-MW (electric) class liquid-metal fast breeder reactor with one-third of the control rods inserted, which may increase to 0.4 to 0.5% for a case where all control rods are withdrawn.

- 3080 NATURAL CIRCULATION BOOSTS SAFETY OF POOL-TYPE FAST REACTOR. Gregory, C.V.; Anderson, R.; Bell, R.; Brown, G.A.; Dawson, C.W.; Lord, D.J. (UKAEA, Dounreay, Nuclear Power Development Establishment). Nucl. Eng. Int.; 24: No. 293, 29-33(Dec 1979).

Experiments are described which were carried out at Dounreay to investigate natural circulation within the PFR primary vessel. If, during a primary pump rundown caused by loss of electrical supplies, all three auxiliary motors were to fail, natural circulation, if demonstrable, could transfer heat from the core to the decay heat rejection loops. The results of the experiments showed that because of the good heat transfer properties of sodium and also the large thermal capacity of the sodium pool, natural circulation is a powerful intrinsic safety feature of pool type LMFBRs. The experiments also resulted in a smooth transition to natural circulation with temperatures which suggested that in the event of loss of all primary circuit pumping power natural circulation will provide inherent protection and coolant and fuel pin temperatures will not exceed normal operating temperatures. (UK).

- 3081 ANNULAR LINEAR INDUCTION PUMP WITH AN EXTERNALLY SUPPORTED DUCT. Craig, E.R.; Semken, R.S. (to Dept. of Energy). US Patent 4,174,190. 13 Nov 1979. Filed date 30 Jun 1977. 8p.

PAT-APPL-811,627.

Several embodiments of an annular linear induction pump for pumping liquid metals are disclosed having the features of generally one pass flow of the liquid metal through the pump and an increased efficiency resulting from the use of thin duct walls to enclose the stator. The stator components of this pump are removable for repair and replacement. 15 claims.

- 3082 SIMPLIFIED INTERASSEMBLY HEAT TRANSFER MODEL FOR THE ANALYSIS OF LIQUID-METAL FAST BREEDER REACTOR CORE RESTRAINT SYSTEMS. Wei, J.P. (General Electric Co., Sunnyvale, CA). Nucl. Technol.; 46: No. 1, 44-52(Nov 1979).

A simplified interassembly heat transfer model has been developed to satisfy liquid-metal fast breeder reactor core restraint system analysis needs that explicitly treats steady-state intra-assembly and interassembly heat transfer in core assemblies. The intra-assembly heat transfer inside reactor assemblies is modeled based on application of the subchannel concept together with the use of bulk parameters for coolant velocity and coolant temperature within a subchannel. The model utilizes a tri-grid system to treat interassembly heat transfer between assemblies. Because of this special nodal scheme, a set of finite difference equations, derived from the energy equation for all the subchannels, duct wall, and gap flow, is actually a rather special system of simultaneous linear algebraic equations which have a tri-diagonal matrix

form. Due to this special form, an efficient method of solution for computers is used without matrix elimination and inversion. Although this model was developed for core restraint applications, it is also well suited for the determination of core-wide coolant temperature distributions.

- 3083 OBSERVATIONS OF DILATION AND BOWING IN EXPERIMENTAL BREEDER REACTOR II DUCTS AND CLADDING. Walters, L.C.; Walter, C.M. (Argonne National Lab., Idaho Falls, ID). Nucl. Technol.; 46: No. 1, 134-148 (Nov 1979).

Irradiation-induced creep and swelling of stainless-steel Experimental Breeder Reactor II (EBR-II) components have been the subjects of intense surveillance programs during the operation of the EBR-II. Dilation, length change, and bow due to creep and swelling of stainless-steel ducts have all been of concern during the operation of the EBR-II, but in no instance have significant reliability or safety problems arisen due to these phenomena. Driver-fuel elements of two different designs have been irradiated to cladding breach. It was demonstrated that metal driver fuel is capable of high burnup when designed for interconnected porosity prior to fuel-cladding contact. Cladding breaches were found either adjacent to the fuel column or in the fuel-restrainer dimpler. The breach mode was intergranular cracking at both locations.

- 3084 BREEDER REACTOR: A VITAL OPTION. Golan, S. (Bechtel Corp., San Francisco, CA). Aware; No. 110, 6-8 (Nov 1979).

Appeared in Bechtel Briefs, September 1979. Plans for the breeder reactor depend more on political than scientific decisions. A breeder produces 60 times more energy from the same uranium resource. Breeder technology was pioneered by the US, but Russia, France, and UK are developing the breeder at the determined pace, while development in the US is stalled. The LMFB is in demonstration and is approaching commercialization. Mr. Golan says that we cannot, nor should be expected to, demonstrate that the breeder is now an essential element in our national energy production program. The breeder is insurance against a major future constraint on energy availability. If the best of circumstances comes about so that the breeder is not so urgently needed as it now appears, we have only made an eventually needed investment 5 or 10 years too early. If the worst of the range of possible circumstances develops, our nation will be grateful that we had the foresight to develop it. Proceeding with the construction of an experimental plant does not commit the US to large-scale deployment of breeder reactors, we are simply developing an energy option. (MCW)

- 3085 SIMULTANEOUS FREEZING AND MELTING IN FINITE MEDIA. Hakim, S.J. (Argonne National Lab., IL). Nucl. Sci. Eng.; 72: No. 2, 129-139 (Nov 1979).

From ASME winter meeting on simultaneous melting and freezing in the impingement region of a liquid jet; San Francisco, CA, USA (Win 1978).

Simultaneous crust growth in a fluid and the melting of an adjacent colder structure surface in contact with it are examined with emphasis on fast breeder reactor safety applications. The calculations were made subject to the following assumptions: (1) As long as a crust exists, melted structure adjacent to it stays in place until its melts. (2) When a crust melts, melted structure is ablated. The ablation process does not influence the

temperature profile in the remaining intact structure. The dependence of the Fourier number at which complete structure melting occurs on fluid-to-crust heat flux is obtained for uranium oxide and thorium oxide crusts forming on steel surfaces. The crust behavior is also investigated as the heat flux is varied. The dependence of the results on internal heat generation and the Biot number on the other side of the structure is determined.

- 3086 PLUTONIUM PROCESSING FOR THE FAST REACTOR FUEL CYCLE. Allardice, R.H. (UKAEA Culham Lab., Abington); Taylor, H.A. Nucl. Eng. Int.; 24: No. 292, 52-55 (Nov 1979).

Current Dounreay and Windscale experience in reprocessing, fabrication and plutonium waste operations together with the continuing UK development programme to establish process and plant design for larger scale operations is described. The necessity for considering these fuel cycle operations as an entity and avoiding the independent optimization of individual areas, in order to obtain the preferred balance for safe and reliable Pu processing operations is emphasized.

- 3087 PLUTONIUM RECOVERY GOES ON-STEAM. Barrett, T.R.; Mills, A.L. Processing; 25: No. 11, 42-43 (Nov 1979).

The Dounreay fast reactors DFR and PFR are very briefly described, with particular reference to their fuel elements. The spent fuel reprocessing plant is described in more detail under the headings: fuel subassemblies (dimensions, composition, preparation for reprocessing); dissolution and recovery (U-Pu separation from fission products, etc., and from each other); waste products.

- 3088 DOUNREAY REPROCESSING PLANT CLOSES NUCLEAR FUEL CYCLE. Spear, M. Process Eng.; 132-133 (Nov 1979).

The first fast reactor fuel reprocessing plant which has been officially opened at Dounreay will, when fully operational, represent our important link to the fast breeder program, reprocessing irradiated fuel from the PFR. A brief outline is given of the PFR fuel assemblies and of the operations following their removal from the core and in the reprocessing plant. Building the plant on the old plant site at a cost of Pound3.4 million, represents a saving of some Pound10 million, compared with the cost of a completely new plant.

- 3089 DOSE MORTALITY RELATIONSHIPS: IMPLICATIONS FOR HYPOTHETICAL ACCIDENTAL RELEASES FROM FBRS. Kelly, G.N. (National Radiological Protection Board, Harwell (UK)). Radiol. Prot. Bull.; No. 31, 16-18 (Nov 1979).

A summary is given of the findings detailed in the most recent of a series of radiological studies of a liquid metal-cooled fast breeder reactor (Kelly, G.N. Simmonds, J.R. Smith, H. and Stather, J.W., The radiological consequences of national accidental releases of radioactivity from fast breeder reactors: sensitivity to the dose-effect relationships adopted for early biological effects, Harwell, National Radiological Protection Board, NRPB-R87. London, HMSO (1979)). The results have indicated those areas where effort might be most profitably directed to improve the reliability of the predicted consequences. The composition of the released activity has been shown to have a marked influence on the sensitivity of the predicted consequences, and application of simple supportive medical

treatment has a potential for reducing the predicted consequences by factors of up to an order of magnitude, depending on the composition of the release.

- 3090 UNDERWRITING STRUCTURAL STEELS FOR THE SODIUM-COOLED FAST REACTOR. Charnock, W. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.); Haigh, C.P.; Horton, C.A.P.; Marshall, P. CEBG Res.; No. 10, 3-14(Nov 1979).

Material strength and fabrication considerations have led to the choice of AISI type 316 stainless steel for most of the components in the primary circuit and the intermediate heat exchanger for the UK fast reactor. High strength chromium-molybdenum ferritic steels are a common choice for boiler tubes and shells of the steam generator units. The physical and chemical influences of liquid sodium upon these structural steels are summarised. Laboratory techniques which have been used in investigating mechanical properties in various sodium conditions are described. Some of the results are presented and future plans for testing are discussed.

- 3091 ALLOYS FOR A LIQUID METAL FAST BREEDER REACTOR. Rowcliffe, A.F.; Bleiberg, M.L.; Diamond, S.; Bajaj, R. (to Dept. of Energy). US Patent 4,172,742. 30 Oct 1979. Filed date 6 Jan 1978. 12p. PAT-APPL-867,656.

An gamma-prime was precipitation-hardened iron-chromium-nickel alloy was designed with emphasis on minimum nickel and chromium contents to reduce the swelling tendencies of these alloys when used in liquid metal fast breeder reactors. The precipitation-hardening components have been designed for phase stability and such residual elements as silicon and boron, also have been selected to minimize swelling. Using the properties of these alloys in one design would result in an increased breeding ratio over 20% cold worked stainless steel, a reference material, of 1.239 to 1.310 and a reduced doubling time from 15.8 to 11.4 years. The gross stoichiometry of the alloying composition comprises from about 0.04% to about 0.06% carbon, from about 0.05% to about 1.0% silicon, up to about 0.1% zirconium, up to about 0.5% vanadium, from about 24% to about 31% nickel, from 8% to about 11% chromium, from about 1.7% to about 3.5% titanium, from about 1.0% to about 1.8% aluminum, from about 0.9% to about 3.7% molybdenum, from about 0.04% to about 0.8% boron, and the balance iron with incidental impurities.

- 3092 PROCESS AND APPARATUS FOR THE DETECTION AND PRELOCALIZATION OF A CAN FRACTURE IN A FUEL ASSEMBLY OF A FAST NUCLEAR REACTOR. Paziaud, A. (to Commissariat a l'Energie Atomique). US Patent 4,172,012. 23 Oct 1979. Priority date 23 Jun 1976, France, 4p.

Process and apparatus for the detection and prelocalization of a can fracture in a fuel assembly of a fast nuclear reactor are described. The device comprises at least one detector for the neutrons emitted by the fission products transported in the outflow of cooling agent as the result of the fracture of a can. The detector being immersed within the vessel in the outflow of cooling agent in an area protected from neutrons emitted by the core by the volume of cooling agent and by structures immersed in the annular space between the core and the vessel.

- 3093 LIQUID METAL COOLED NUCLEAR REACTOR

CONSTRUCTIONS. Seed, G.; Hodgson, D.; Grime, C.J. (to Nuclear Power Co. Ltd.). US Patent 4,172,010. 23 Oct 1979. Priority date 12 Aug 1976, United Kingdom of Great Britain and Northern Ireland (UK), 4p.

The fuel assembly in a nuclear reactor construction is supported from the wall of a primary vessel on a strongback with a secondary support means from the roof structure of a containment vault by ties. The secondary support provides a safety device whereby in the event of relaxation of the primary support relative movement between the fuel assembly and control rods is limited. Alternative secondary support means comprise a series of ties supporting the diagrid and a support stool beneath the fuel assembly.

- 3094 NUCLEAR REACTOR COMPOSITE FUEL ASSEMBLY. Burgess, D.M.; Cappiello, M.W.; Marr, D.R.; Omberg, R.P. (to Westinghouse Electric Corp., Pittsburgh, PA; USA). British Patent 2,018,009/A/. 10 Oct 1979. 8p.

A fuel assembly and core for a liquid-cooled breeder reactor is described which, it is claimed, reduces deficiencies found in typical present designs while maintaining a relatively high breeding ratio and low doubling time. (UK).

- 3095 LIQUID METAL COOLED FAST BREEDER NUCLEAR REACTORS. Gatley, J.A. (to Nuclear Power Co. Ltd., London; UK). British Patent 2,017,385/A/. 3 Oct 1979. 6p.

Breeder fuel sub-assemblies with electromagnetic brakes and fluidic valves for liquid metal cooled fast breeder reactors are described. The electromagnetic brakes are of relatively small proportions and the valves are of the controlled vortex type. The outlet coolant temperature of at least some of the breeder sub-assemblies are maintained by these means substantially constant throughout the life of the fuel assembly without severely pressurising the sub-assembly. (UK).

- 3096 HIGH INTEGRITY STEAM GENERATOR TUBE/TUBE PLATE JOINTS FOR THE LMFBR. Taylor, A.F. (Springfields Nuclear Power Development Labs., Salwick, England); Hayden, D. Weld. J. (Miami); 59: No. 10, 31-37(Oct 1979).

An extensive, two-year development program is now coming to a close. The basic process details necessary to obtain high integrity brazed joints for heat exchanger applications have been defined and are now being verified on a production basis by a leading UK Boilermaker. Replacement tube bundles are currently in manufacture and will be installed in the Prototype Fast Reactor at Dounreay in Scotland during 1981. The use of brazed joints in a sodium-heated steam generator may appear an exacting requirement. However, a comprehensive evaluation program has indicated that they will meet the stress and corrosion conditions anticipated in service. Confidence gained during the testing program has led to a further application at present under development. This is to exploit the technique in the repair of leaking tube-to-tube plate welds in current heat exchangers; the procedures being developed could have application to heat exchanger designs involving closely pitched smaller bore tubes.

- 3097 ANALYTIC PREDICTION OF THERMAL PROFILE DEVELOPMENT AND INVERSION FOR USE IN IN-PILE EXPERIMENTATION. Weber, D.P. (Argonne National Lab., IL). Nucl. Technol.; 45: No. 3, 203-218(Oct 1979).

The assessment of the consequences of hypothetical accidents in liquid-metal-cooled fast reactors often requires interaction between analysis and in-pile experiments, where experiments must provide geometry, boundary conditions, and thermal profiles that are prototypical of the accident scenario. Neutronic heating of test samples initially produces atypical thermal profiles, and a time period is required to elapse for thermal inversion. An analytic transient heat conduction analysis using multiregion eigenfunctions is provided to determine the space-time temperature profiles. With an assumed weak temporal dependence for eigenfunctions greater than the first, a determination of the motion of the position of maximum temperature is made, leading to a simple expression for the time to thermally invert completely, which requires knowledge of only the first eigenvalue and the expansion coefficient of the source for the fundamental mode, with similar analysis providing an estimate of the time to reach melting. A functional relationship is established between the operating reactor power, the thermal properties of the materials, and the boundary conditions to ensure satisfaction of both criteria of rapid thermal inversion and maximum temperatures above prescribed levels, such as melting. The analysis is then applied to a proposed in-pile experiment for studying pool boilup in internally heated fuel-steel pools with nuclear heated walls. It is shown that for a variety of external boundary conditions, a reactor power level may be chosen to ensure integrity of the insulating wall while simulating the pool boilup phenomena without the necessity of enrichment grading to enhance thermal inversion.

- 3098 TREAT TEST H6-A 50 CENTS TRANSIENT OVERPOWER ACCIDENT SIMULATION. Page, R.J.; Fink, C.L.; Rothman, A.B.; Lo, R.K.; Robinson, L.E.; Froehle, P.H. (Argonne National Lab., IL). Nucl. Technol.; 45: No. 3, 249-268(Oct 1979).

Transient Reactor Test Facility (TREAT) Test H6 was run to simulate a transient overpower (TOP) initiated 50 cent/s hypothetical core disruptive accident (CDA). The primary purpose was to investigate the extent to which molten fuel could be removed from the active core region following fuel pin failure, and the extent to which this would be accomplished while maintaining coolant flow. Coolant flow rates and pressures typical of those in the Fast Flux Test Facility were used. The test fuel sample consisted of a bundle of seven mixed-oxide fuel pins which had been preirradiated in the Experimental Breeder Reactor II to approx. 6 at % burnup. The liquid sodium coolant had an initial velocity of 6.20 m/s at a temperature of 742 K. A programmed TREAT power ramp with a period of 1.65 s was used to bring the experimental fuel sample to failure conditions. The test data showed that there were three main events associated with fuel pin failure. During the first of these events, fuel was removed from the active fuel region and relocated approx. 40 cm downstream. The coolant flow rate recovered to approx. 93% of its preevent value. Additional fuel was removed from the active fuel region during the second event and again relocated some 40 cm downstream. However, molten fuel also began to accumulate in a region centered on the centerline of the original fuel column. The coolant flow rate recovered to approx. 75% of its initial value. The third event was considerably more violent than the others and while a considerable quantity of fuel was relocated well downstream of the active fuel

column, a blockage was formed at the top of the fuel column which reduced the coolant flow to zero. Analysis of the flowmeter signals indicated that the fuel pin holder failed during the third event. This could be at least partially responsible for the coolant channel blockage following this event.

- 3099 BOR-60 SODIUM BOILING EXPERIMENT. Afanas'ev, V.A.; Aleksandrov, K.A.; Kebabze, B.V. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)); Fuge, R.; Kaetzer, D. (Zentralinstitut fuer Kernforschung, Rossendorf bei Dresden (German Democratic Republic)). Kernenergie; 22: No. 10, 357-363(Oct 1979).

Sodium boiling was generated in the BOR-60 core by γ -heating of tungsten rod bundles in two different test rigs. The boiling noise was measured by means of an ionization chamber and several acoustic transducers. The results of noise analysis of the background and the boiling noise are presented and discussed from the point of view of reactor surveillance.

- 3100 CONTRIBUTION OF DOUNREAY TO THE DEVELOPMENT OF FAST REACTORS. Blumfield, C.W. (UKAEA Dounreay Nuclear Power Development Establishment). Nucl. Energy; 18: No. 5, 311-326(Oct 1979).

A review is presented of the fast reactor work performed at Dounreay on fuel and core materials, the reactor system, and fuel reprocessing. The following topics are discussed: Dounreay's role in the development work on ceramic fuels leading to a specification for PFR fuel; irradiation behaviour of materials, eg., quantification of effects of irradiation in mechanical properties, irradiation creep of stainless steels and work on void swelling; monitoring of reactor components and coolant; investigation of the performance of highly rated fuel under adverse conditions; and testing of oxide fuel elements in PFR. The determination of the behaviour of PFR boiler tubes and the study of the features large leaks, including the sodium-water reaction have been investigated in the large Super-Noah rig at Dounreay are described. Finally development work for PFR fuel reprocessing is discussed. This includes the dismantling of fuel assemblies, fuel dissolution, volume reduction and segregation of contaminated waste, and treatment of radioactive waste.

- 3101 CLOSING THE FAST FUEL CYCLE AT DOUNREAY. Moss, J. Electr. Rev. (London); 205: No. 11, 16-17(21 Sep 1979).

The opening of the new fast reactor fuel reprocessing plant at Dounreay enables plutonium based fuel used in the PFR to be reprocessed and the plutonium to be sent to Windscale for fabricating into new fuel rods. An outline is given of the fuel cycle including plans for waste disposal.

- 3102 NUCLEAR REACTOR. Aubert, M.; Lambert, P. (to Commissariat a l'Energie Atomique). US Patent 4,167,445. 11 Sep 1979. Priority date 26 Nov 1975, France, 10p.

A first annular space is formed between a cylindrical baffle and the reactor vessel and a second annular space is formed between the baffle and a counter-baffle. The level of liquid metal coolant is free within the two annular spaces which are separated by the baffle. The top edge of the baffle has a profile such that the supply of cold liquid metal which overflows into the discharge space

takes place without breaking away from the baffle wall.

- 3103 SELF-ACTUATED RATE OF CHANGE OF PRESSURE SCRAM DEVICE FOR NUCLEAR REACTORS. Noyes, R.C.; Zaman, S.U.; Stuteville, D.W. (to Combustion Engineering, Inc.). US Patent 4,167,443. 11 Sep 1979. Filed date 30 Aug 1977. 6p.

A sensor chamber having one cavity containing coolant separated by a diaphragm from another cavity containing a fixed mass of inert gas is located within a safety assembly of a liquid metal-cooled nuclear reactor. The liquid cavity is in fluid communication with the coolant outside the chamber through a flow limiting orifice. An actuating bellows in fluid communication with the gas cavity is in contact with coolant outside the chamber and is connected to a push rod, which serves as a trigger for a poison bundle release mechanism. During slow changes in reactor coolant pressure experienced under normal operation, the diaphragm moves to equalize the gas cavity and liquid cavity pressures with the coolant pressure outside the chamber. The actuating bellows does not move because it is biased so that a threshold pressure difference is required before it will expand. Under a more rapid drop in coolant pressure, such as is associated with a loss of forced flow, the threshold is overcome and the actuating bellows will also move, thereby triggering the release mechanism to shut down the reactor. The actuating bellows may be connected to the liquid cavity rather than to the gas cavity.

- 3104 HIGH TEMPERATURE TESTING OF A SODIUM PUMP. Atz, R.W.; Tessier, M.J. Contract EY-76-C-03-0700. New York; American Society of Mechanical Engineers (Sep 1979). 8p.

Problems encountered during high temperature sodium testing of a prototype breeder reactor heat transport pump at the Energy Technology Engineering Center (ETEC) are discussed. Special tests and analyses to resolve the uneven heating and resulting high shaft torque which was traced to cellular natural convection is discussed. Vibrations were observed during steady-state testing. Diagnostic testing and analyses (e.g., power spectral density) were developed to determine the source and to effect a solution. The source was discovered to be vortex shedding off the upper impeller wear ring outlet; subsequent pump modification, by stiffening the ring, mitigated the problem. During thermal transient testing, a shaft seizure by the hydrostatic bearing occurred. Densification of the bearing support material during prolonged high temperature operation caused the loss of clearance.

- 3105 MECHANISM OF RELEASE OF RADIOACTIVE PRODUCTS INTO LIQUID-METAL COOLANTS, THEIR TRANSPORT WITHIN THE CIRCUITS AND REMOVAL FROM LMFBRs. Feuerstein, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)); Hooper, A.J. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.); Johnson, F.A. (UKAEA, Harwell. Atomic Energy Research Establishment). At. Energy Rev.; 17: No. 3, 697-761 (Sep 1979).

The sources, distribution and removal of the radioisotopes contributing to LMFBR primary circuit contamination are considered in three separate sections. The principal mechanisms of fission product generation and release from fast reactor fuels during normal reactor operation are examined. Published data on the radioactivity in circuits of experimental and power reactors, and for loop experiments, are

assessed and compared. Information on the behaviour of individual isotopes in the coolant circuit and cover gas and their removal via cold traps and other specialized means are reviewed.

- 3106 (DOE-tr--115) ANALYSIS OF THE DISASSEMBLY PHASE OF HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS (HCDA) FOR THE SNR-2000. Maschek, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Sep 1979. Translation of KFK--2491, December 1977 202p. Dep. NTIS (US Sales Only), PC A10/MF A01.

Work performed under United States-West Germany Fast Reactor Exchange Program.

During 1976 a first safety analysis of a large liquid metal fast breeder reactor (LMFBR) - the so-called SNR-2000 - has been performed in the framework of the fast breeder project. The reactor produces a thermal power of 5000 MW corresponding to 2000 MW electric. The analysis mainly dealt with the behavior of HCDAs in the SNR-2000 core. Transient over power (TOP), loss of flow (LOF), and LOF driven TOP's have been analyzed for two burn up status (begin of equilibrium cycle A (BEC-A) and end of equilibrium cycle B (BEC-B)). For these type of accidents the failure of the redundant and diversive shut down system has been assumed.

- 3107 MEASUREMENTS OF HYDROGEN CONCENTRATION IN LIQUID SODIUM BY USING AN INERT GAS CARRIER METHOD. Funada, T.; Nihei, I.; Yuhara, S.; Nakasugi, T. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki, Japan). Nucl. Technol.; 45: No. 2, 158-165 (Sep 1979).

A technique was developed to measure the hydrogen level in liquid sodium using an inert gas carrier method. Hydrogen was extracted into an inert gas from sodium through a thin nickel membrane in the form of a helically wound tube. The amount of hydrogen in the inert gas was analyzed by gas chromatography. The present method is unique in that it can be used over the wide range of sodium temperatures (150 to 700°C) and has no problems associated with vacuum systems. The partial pressure of hydrogen in sodium was determined as a function of cold-trap temperature ($T_{\text{sub c}}$). Sieverts' constant ($K_{\text{sub s}}$) was determined as a function of sodium temperature (T). From Sieverts' constant, the solubility of hydrogen in sodium is calculated. It was found that other impurities in sodium, such as (D) and (OH), have little effect on the hydrogen pressure in the sodium loop.

- 3108 PLANNING FOR THE FUTURE GENERATION. Heat. Air Cond. J.; 49: No. 571, 18-26 (Sep 1979).

The UK nuclear power programme is discussed under the following headings: introduction; reactor types (British commercial reactors - history, design, the Magnox reactors, the AGR, reactors in development - PFR); the cost of nuclear power; the nuclear fuel cycle; safety; organisation in the UK; fusion; the nuclear energy contribution; assessment.

- 3109 ACCIDENT PROGRESSION FOR A LOSS-OF-HEAT-SINK WITH SCRAM IN A LIQUID-METAL FAST BREEDER REACTOR. Bari, R.A.; Ludwig, H.; Pratt, W.F.; Sun, Y.H. (Brookhaven National Lab., Upton, NY). Nucl. Technol.; 44: No. 3, 357-380 (Aug 1979).

An analysis of a slow core meltdown in a liquid-metal fast breeder reactor was performed for the conditions of loss-of-heat-sink

following neutronic shutdown. Simple models were developed for the prediction of phase changes and/or relocation of the core materials, including fuel, coolant, cladding ducts, control rod absorber material (B₄C), and plenum gases. The sequence of events was accounted for, and the accident progression was described up to the point of recriticality. The neutronic behavior of the disrupted core was analyzed in r - z geometry with a static transport theory code (TWOTRAN). For most scenarios assessed, the reactor is expected to become recritical, although large ramp rates are not anticipated.

- 3110 EFFECT OF FLUID-TO-STRUCTURE HEAT TRANSFER ON THE STRUCTURAL DAMAGE POTENTIAL TO A LIQUID-METAL FAST BREEDER REACTOR. Hakim, S.J.; Abramson, P.B. (Argonne National Lab., Argonne, IL). Nucl. Technol.; 44: No. 3, 390-400 (Aug 1979).

Deterministic calculations simulating a hypothetical accident in a liquid-metal fast breeder reactor that leads to a hydrodynamic disassembly of the core have been carried out to estimate the system's damage potential due to the vapor-pressure-driven expansion of molten core material and its dependency on the heat transfer to the remaining structure. These calculations ignored the effect on the work potential of sodium left in the core during the disassembly. Results indicate that steel cladding in the upper axial blankets and fission gas plenum acts as a thermodynamic energy sink that could reduce the total thermodynamic work energy by between one and two orders of magnitude, provided little or no sodium is present in the core at the time of interaction. These results have been found to be insensitive to the rate of heat transferred from the molten fuel to the molten steel that comprises the molten core material.

- 3111 ACCURACY AND COMPUTATIONAL TIME OF A HIERARCHY OF GROWTH RATE DEFINITIONS FOR BREEDER REACTOR FUEL. Maudlin, P.J.; Borg, R.C.; Ott, K.D. (Purdue Univ., West Lafayette, IN). Nucl. Sci. Eng.; 71: No. 2, 202-227 (Aug 1979).

For a hierarchy of four logically different definitions for calculating the asymptotic growth of fast breeder reactor fuel, an investigation is performed concerning the comparative accuracy and computational effort associated with each definition. The definition based on detailed calculation of the accumulating fuel in an expanding park of reactors asymptotically yields the most accurate value of the infinite time growth rate, $\gamma/\sup \text{infinity}$, which is used as a reference value. The computational effort involved with the park definition is very large. The definition based on the single reactor calculation of the equilibrium surplus production rate and fuel inventory gives a value for $\gamma/\sup \text{infinity}$ of comparable accuracy to the park definition and uses significantly less central processor unit (CPU) time. The third definition is based on a continuous treatment of the reactor fuel cycle for a single reactor and gives a value for $\gamma/\sup \text{infinity}$ that accurately approximates the second definition. The continuous definition requires very little CPU time. The fourth definition employs the isotopic breeding worths, $w/\sub i/\sup *$, for a projection of the asymptotic growth rate. The CPU time involved in this definition is practically nil if its calculation is based on the few-cycle depletion calculation normally performed for core design and critical enrichment evaluations. The small inaccuracy (approx. =

1%) of the breeding-worth-based definition is well within the inaccuracy range that results unavoidably from other sources such as nuclear cross sections, group constants, and flux calculations. This fully justifies the use of this approach in routine calculations.

- 3112 (DOE-tr-142) POSSIBILITIES FOR THE DETERMINATION OF THE HEAT TRANSFER PROCESS BETWEEN NUCLEAR FUEL RODS AND THE COOLING MEDIUM IN A SODIUM-COOLED REACTOR DURING OPERATION. Mitzel, F. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Aug 1979. Translation of KFK-2619, June 1978 52p. Dep. NTIS (US Sales Only), PC A04/MF A01.

Work performed under United States-W. Germany Fast Reactor Exchange Program.

In order to improve the safety, the surveillance and economy of sodium cooled reactors it is desirable to obtain a more precise knowledge of the heat transfer between the fuel and the coolant. Different possibilities have been investigated to determine the corresponding heat transfer coefficient by measuring the pertinent transfer functions. The result was the following: the heat transfer coefficient can be determined with reasonable accuracy at the KNK II and the SNR by means of small pseudo-stochastic power oscillations. These power oscillations must be artificially induced but do not require an interruption of the normal plant operation.

- 3113 IMPRESSIVE SUPER PHENIX. Olds, F.C. Power Eng.; 83: No. 8, 62-66 (Aug 1979).

The 1200-MWe fast breeder reactor, Super Phenix at Creys-Malville, is scheduled for commercial operation in 1983. This is the world's first near-commercial-sized fast breeder. As a near-commercial-sized unit, it represents essentially the technology and hardware of the first fully commercial follow-on units. In its size, its components, its design, the technology it represents, and its project schedule, it is impressive. As of May 1979, the Super Phenix nuclear steam boiler in the Creys-Malville plant bore an estimated cost of \$700 million, without fuel. The total cost of the Creys-Malville plant now is estimated at about \$1.4 billion. This is about twice the cost of a comparable standardized PWR being built in France today. However, it should be borne in mind that Creys-Malville carries the high cost of a first-of-the-line prototype, and that France's PWRs are standardized, second-generation units. Electricity from Creys-Malville is estimated to cost a little more than electricity would cost from a coal-fired plant complete with flue gas scrubbing.

- 3114 GAS BUBBLE ADHESION TO HEAT TRANSFER SURFACES. Winterton, R.H.S.; Chevalier, A.B.H.; Fowles, G.A. (Birmingham Univ. (UK). Dept. of Mechanical Engineering). Nucl. Eng. Des.; 53: No. 3, 347-354 (Aug 1979).

A model of gas bubble adhesion to heat transfer surfaces is developed, by considering the drag, surface tension and buoyancy forces acting on a gas bubble attached to a solid wall with liquid flowing past. The model is consistent with the fuel pin failures that were observed in the irradiation tests of fuel sub-assemblies for the British Prototype Fast Reactor, and predicts that changing to upward flow will not in itself necessarily prevent such failures in the future.

- 3115 HEAVY SECTION STEEL TECHNOLOGY (HSST)

PROGRAM. Canonico, D.A. (Oak Ridge National Lab., TN). Met. Prog.; 116: No. 2, 32-41 (Jul 1979).

Material characterization, fracture mechanics, fatigue crack growth, pressure vessel tests, irradiation effects, and thermal shock of BWR and PWR steam generators are part of the function of the heavy section steel technology program (HSST).

3116 MATERIALS TECHNOLOGY FOR LMFBR STEAM GENERATORS. Brinkman, C.R. (Oak Ridge National Lab., TN); Katcher, M. Met. Prog.; 116: No. 2, 54-61 (Jul 1979).

Research in support of steam generators for LMFBR reactors is summarized. A description of the reactor development and characterization of alloy fabrication are described.

3117 EXPERIMENTAL STUDY OF SCRAM TRANSIENTS IN GENERALIZED LIQUID-METAL FAST BREEDER REACTOR OUTLET PLENUMS. Howard, P.A.; Carbajo, J.J. (Argonne National Lab., IL). Nucl. Technol.; 44: No. 2, 210-220 (Jul 1979).

An experimental study was made of the transient behavior of the coolant in a generalized outlet plenum after a scram in a liquid-metal fast breeder reactor. Different conditions of fluid flow rate and temperature and plenum geometry were used for a three-exit loop-type reactor system. Water was used to simulate the coolant, and a small-scale model was used to simulate the various generalized plenum configurations. Thermocouples monitored the temperatures at 48 locations within the outlet plenum. The thermohydraulic behavior within the plenum has been studied in detail; analysis of this behavior is described as having three consecutive discrete stages. Correlations found in the literature accurately predict the height of the negatively buoyant plume. New correlations have been obtained.

3118 CARTER'S BREEDER POLICY HAS FAILED, CLAIMS WESTINGHOUSE MANAGER. Electr. Light Power (Boston); 57: No. 7, 26 (Jul 1979).

Nuclear nations developing liquid metal fast breeder reactor (LMFBR) technology have not been dissuaded by President Carter's efforts to stop the breeder program as a way to control the proliferation of nuclear weapons. There is no evidence that Carter's policy of moral persuasion has had any impact on their efforts. A review of the eight leading countries cites their extensive progress in the areas of breeder technology and fuel reprocessing, while the US has made only slight gains. The Fast Flux Test Facility at Hanford is near completion, but the Clinch River project has been slowed to a minimum.

3119 (ANL-Trans--1177) REPORT OF STUDY ON SAFETY WITH RESPECT TO SODIUM WATER REACTION. SECTION 4. WATER LEAK RATE EVALUATION IN THE LARGE SCALE SODIUM WATER REACTION. SECTION 5. CONCLUDING REMARKS. APPENDIX. ANALYSIS CODE: SUMMARY. (Japan Society of Mechanical Engineers, Tokyo). Jul 1979. Translation of Japanese report, March 1978, pp 143-261 175p. Dep. NTIS, PC A08/MF A01.

Methods for the estimation of water leak rate in an experiment on large-scale sodium water reactions are evaluated.

3120 SUCCESS AND PROBLEMS OF FAST BREEDER DEVELOPMENT. REPORT ON THE 1979 STATUS REPORT OF THE FAST BREEDER PROJECT GIVEN IN KARLSRUHE ON FEBRUARY 12, 1979. Hueper, R. (Kernforschungszentrum Karlsruhe G.m.b.H.

(Germany, F.R.)). Atomwirtsch., Atomtech.; 24: No. 7, 385-386 (Jul 1979). (In German).

After a pause of 3 years, a status report of the Fast Breeder Project was given at Karlsruhe Nuclear Research Centre on February 12, 1979. The report dealt mainly with the construction and operation of the SNR-300 reactor; it gave an outline of the political situation of the fast breeder reactor, German-French cooperation, nuclear safety in large breeders, and the proliferation safety of their fuel cycle.

3121 FATIGUE-CRACK PROPAGATION RESPONSE OF TWO NICKEL-BASE ALLOYS IN A LIQUID SODIUM ENVIRONMENT. Mills, W.J.; James, L.A. (Hanford Engineering Development Lab., Richland, WA). Contract EY-76-C-14-2170. J. Eng. Mater. Technol.; 101: No. 3, 205-213 (Jul 1979).

The elevated temperature fatigue-crack propagation response of Inconel 600 and Inconel 718 was characterized within a linear-elastic fracture mechanics framework in air and low-oxygen liquid sodium environments. The crack growth rates of both nickel-base alloys tested in liquid sodium were found to be considerably lower than those obtained in air. This enhanced fatigue resistance in sodium was attributed to the very low oxygen content in the inert sodium environment. Electron fractographic examination of the Inconel 600 and Inconel 718 fatigue fracture surfaces revealed that operative crack growth mechanisms were dependent on the prevailing stress intensity level. Under low growth rate conditions, Inconel 600 and Inconel 718 fracture surfaces exhibited a faceted, crystallographic morphology in both air and sodium environments. In the higher growth rate regime, fatigue striations were observed; however, striations formed in sodium were rather ill-defined. These indistinct striations were attributed to the absence of oxygen in the liquid sodium environment. Striation spacing measurements were found to be in excellent agreement with macroscopic growth rates in both environments.

3122 FATIGUE CRACK GROWTH BEHAVIOR OF STAINLESS STEEL TYPE 316 PLATE AND 16-8-2 WELDMENTS IN AIR AND HIGH-CARBON LIQUID SODIUM. Yuen, J.L.; Copeland, J.F. (General Electric Co., San Jose, CA). Contract EY-76-C-03-0893. J. Eng. Mater. Technol.; 101: No. 3, 214-223 (Jul 1979).

The fatigue crack growth rates of Type 316 stainless steel and 16-8-2 gas tungsten arc welds were measured in air and high-carbon liquid sodium at 510°C (950°F) for secondary sodium loop Liquid Metal Fast Breeder Reactor applications. Loading variables such as low frequency, load ratio, and hold time at peak load were studied. Supporting studies show that carburization of crack faces occurs prior to any crack tip carburization for the crack sizes and load frequencies of this study. Thus, a carburization effect on da/dN is not shown for relatively short-term tests. Low stress intensity, ultra-low frequency tests to simulate service would be required to show this effect. The crack growth rate curve for 16-8-2 welds was below Type 316 base metal curves for both air and high-carbon sodium at 510°C (950°F). Crack growth rate curves for Type 316 and 16-8-2 welds in high-carbon sodium were below the corresponding curves for air.

3123 MECHANISTIC ANALYSIS OF LMFBR ACCIDENT ENERGETICS. Boudreau, J.E. (Los Alamos Scientific Lab., NM). Nucl. Saf.; 20: No. 4, 402-413 (Jul 1979).

The state of the art is reviewed for liquid-metal-cooled fast breeder reactor (LMFBR) postdisassembly energetics analysis. Previous attempts have provided bounding and conservative efficiency estimates for converting fission energy into primary-system damage. However, calculational results using the SIMMER-II code indicate that current US primary-system designs may withstand substantially larger reactivity insertion rates than previously thought. Detailed results are presented for the energetics resulting from a voided-core expansion, along with a discussion of code-verification activities. Further results are presented for expansions involving other initial and boundary conditions, and the importance of the sodium initially present in the core is defined. Finally, the range of reactivity insertion rates that the primary system can accommodate is estimated for the voided-core case, and the remaining technical issues are defined.

- 3124 INTERNATIONAL PROGRESS WITH LIQUID-METAL FAST REACTORS. Moore, J. (UKAEA, Risley. Nuclear Power Development Establishment). Nucl. Eng. Int.; 24: No. 287, 37-49(Jul 1979).

The main lessons to be learnt from the operation of existing prototype fast reactors, ie, PFR, Phenix and BN350, are summarised. Experience has been good and it is expected that costs can be reduced. World energy requirements and uranium requirements for the next 50 years are examined. Uranium resources are limited and FBRs offer better uranium utilisation. The environment issues and risks of proliferation associated with fast reactor development are discussed. Safety and licensing aspects are considered. Comparisons are made between the two different basic design concepts, pool and loop types. The importance of fuel cycle performance is stressed. Finally programmes on fast reactor work in the European countries, Japan, Russia and USSR are summarised.

- 3125 TREAT: 20 YEARS OF TRANSIENT TESTING. Allen, A. (Argonne National Lab., IL (USA)). Nucl. Eng. Int.; 24: No. 287, 50-51(Jul 1979).

The Transient Reactor Test Facility (TREAT), after 20 years of operating experience, is scheduled for updating during the next three years in order to provide a new advanced test loop which will allow more realistic transient testing of LMFBR fuel bundles. Reasons for the past success of the facility are briefly discussed. Past and future programmes are considered.

- 3126 NUCLEAR REACTOR SAFETY. Head, J.L.; Goddard, A.J.H.; Grant, P.J. Phys. Rep.; 53: No. 4, 249-339(Jul 1979).

An account is given of the basic physics and engineering underlying the design of nuclear reactors, descriptions of the types of reactor currently operating, the principles of reactor fault detection and control, safety analysis and hazards assessment. The more serious hypothetical accidents are reviewed for water cooled and gas cooled reactors and for the liquid metal cooled fast breeder reactor. Safety criteria and the siting of nuclear power stations are discussed and illustrated by reference to procedures in the U.K. and a short examination is made of the environmental effects of nuclear power, spent fuel re-processing and the storage of highly radioactive waste.

- 3127 OPTIMUM NUCLEAR PIN BUNDLE FOR

PROTOTYPIC THERMAL-HYDRAULIC ENVIRONMENT IN LARGE-SCALE TESTING. Nguyen, D.H. (Sandia Labs., Albuquerque, NM (USA)). Nucl. Eng. Des.; 53: No. 2, 207-215(Jul 1979).

The optimum number of fuel pins in a test vehicle to achieve prototypic thermal-hydraulic conditions at the onset of pin failure has been determined. The power history considered was a prompt burst superposed on a low steady state power level. Slower transients without pre-clad-failure boiling did not change the conclusions presented. The analysis used the computer program COBRA-IV-I, with a unique capability of large wire-wrapped asymmetric rod bundles up to 217 pins. The results indicate that a 61-pin test scale is desirable to achieve thermal-hydraulic conditions for prototypic testing, with 37 pins probably an acceptable minimum.

- 3128 POTENTIAL FOR PLUGGED CORE PRESSURIZATION FOLLOWING A HYPOTHETICAL ACCIDENT IN AN LMFBR. Rothrock, R.A.; Kazimi, M.S. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). Nucl. Eng. Des.; 53: No. 2, 197-205(Jul 1979).

Steady state and transient conditions for a two-phase liquid-vapor cylindrical UO_2 pool are analytically investigated. The analysis relies on extrapolation of experimental results from internally heated boiling simulant fluid pools. The internal heat generation rates that can sustain a vapor-liquid pool under steady state conditions are calculated. The pressure transients that can be expected in the event of an imbalance between heat generation rate and heat removal rate are also calculated. The analysis presented demonstrates the significance of accounting for heat removal from the pool for proper assessment of its pressurization potential.

- 3129 BUILDING OF THE FIRST COMMERCIAL FAST BREEDER IN FRANCE. Odoewskii, M.S. Arch. Energiewirtsch.; 33: No. 7, 554-565(Jul 1979). (In German).

A survey is given on the technical particularities of the super phenix reactor which is to acquire its full power in 1985. The building of this reactor makes use of the experience gained with the Phenix pilot reactor. The state of building in March 1978 is reported.

- 3130 LIQUID METAL COOLED FAST BREEDER NUCLEAR REACTORS. Duncombe, E.; Thatcher, G. (to UKAEA London Office). British Patent 2,010,000/A/. 20 Jun 1979. 6p.p.

The invention described relates to a liquid metal cooled fast breeder nuclear reactor in which the fuel assembly has an inner zone comprised mainly of fissile material and a surrounding outer zone comprised mainly of breeder material. According to the invention the sub-assemblies in the outer zone include electro-magnetic braking devices (magnets, pole pieces and armature) for regulating the flow of coolant through the sub-assemblies. The magnetic fields of the electro-magnetic braking devices are temperature sensitive so that as the power output of the breeder sub-assemblies increases the electro-magnetic resistance to coolant flow is reduced thereby maintaining the temperature of the coolant outlets from the sub-assemblies substantially constant.

- 3131 METHOD AND APPARATUS FOR CONTROLLING THE NEUTRON FLUX IN NUCLEAR REACTORS. Minnick, L.E. (to Electric Power Research Institute, Inc.). US Patent 4,158,602. 19 Jun 1979.

Filed date 27 Jan 1977. 12p.

A control rod assembly in a nuclear reactor that automatically scrams the reactor when a loss of coolant flow occurs and that can also control the level of neutron flux in the reactor is described. The control rod assembly includes a separator plate having an orifice through which the reactor coolant flows and a sealing surface around the orifice. The control rod in the assembly has a complementary sealing surface. When the control rod and separator plate are brought into contact, the differential pressure across the separator plate caused by the flow of the primary coolant through the reactor core retains the two sealing surfaces together. If the flow of coolant stops or the differential pressure across the separator plate decreases for any reason, the control rod drops by gravity and the reactor is scrammed. The control rod is also automatically dropped as a result of the lateral vibration of an earthquake or by the downward motion of the rod drive shaft, either of which will open the sealing surfaces and reduce the sealing pressure.

- 3132 DEVICE FOR MEASURING NEUTRON FLUX IN A NUCLEAR REACTOR. Marmonier, P.L.; Moreau, J.; Nervi, J.C. (to Commissariat a l'Energie Atomique). US Patent 4,157,277. 5 Jun 1979. Priority date 21 Sep 1976, France, 8p.

A neutron guide consisting of two superposed and separable tubular elements extends from the reactor core to the immediate vicinity of the bottom wall of the reactor vessel directly above a neutron detection chamber located outside the vessel. The upper element contains a neutron-emissive medium a closed tube filled with inert gas engaged coaxially with the lower element which is positioned within a support sleeve. The lower element is provided with another closed sleeve filled with inert gas and terminating near the internal surface of the bottom of the reactor vessel.

- 3133 WEST GERMAN PERSPECTIVE ON THE NEED FOR THE PLUTONIUM-FUELED LMFBR. Hennies, H.H.; Jansen, P.; Kessler, G. Nucl. News; 22: No. 8, 69-75 (Jun 1979).

The need for the deployment of fast breeder reactors in West Germany is discussed and some of the arguments against deployment are examined. Topics covered include future energy demand, uranium demand in West Germany, world uranium demand, proliferation aspects, weapons use of nuclear materials, amount of sensitive materials in the fuel cycle, and alternative fuel cycles.

- 3134 ACOUSTIC BACKGROUND NOISE OF A LIQUID METAL FAST BREEDER REACTOR. Carey, W.M. (Scientific Consultant, Essex, Connecticut 06426). J. Acoust. Soc. Am.; 65: No. 6, 1461-1466 (Jun 1979).

Absolute measurements of the background noise in the upper plenum of an operating liquid metal fast breeder reactor (LMFBR), the Experimental Breeder Reactor II (EBR-II), have been performed over a 17-month period. The temporal spectral density characteristics of the data were found to be representative of a weakly stationary random process. Cross-power spectral density analyses established that below 5 kHz the spectral characteristics were dominated by pump-induced structural vibrations and the excitation of acoustic resonances in the upper plenum volume. The spectral characteristic of the data above 5 kHz were found to be consistent with sonic power generation at the reactor subassembly exit nozzles and the statistical response of the

cylindrical plenum volume. The production of sound by the exit flow from the subassembly nozzles was found to produce a broadband pressure level of 161 dB re 1 μ Pa, corresponding to an acoustic efficiency of 6×10^{-8} . This compares favorably with a theoretical (Lighthill) value of 159 dB re 1 μ Pa and an acoustic efficiency of 9×10^{-9} for a submerged water jet. These measurements show that the noise field in the plenum of a fast reactor is a weakly stationary random process with identifiable sources of sound.

- 3135 NUCLEAR DESIGN AND ECONOMIC COMPARISON OF A CONVENTIONAL AND BULLSEYE LIQUID-METAL FAST BREEDER REACTOR CORE. Vitti, J.A.; Felten, L.D.; Galluzzo, N.G.; Otter, J.M.; Brittingham, J.C. (Atomics International Div., Canoga Park, CA). Nucl. Technol.; 44: No. 1, 10-20 (Jun 1979).

Reactor system design and safety studies at Atomics International have focused on the selection of a core configuration for the Prototype Large Breeder Reactor that has inherent nonenergetic responses to postulated core disruptive accidents. These studies have led to the selection of a non-conventional heterogeneous design, referred to as the bullseye core, as the preferred concept. The nuclear design characteristics and economics of this bullseye core are compared to those of an optimized, conventional liquid-metal fast breeder reactor core configuration, referred to as the regular core. A reduced sodium void worth and an enhanced voiding incoherence are attained by the introduction of internal blanket regions in the bullseye core. The bullseye core total energy costs are shown to be acceptable considering the significant improvement in licensability due to reduced core energetics.

- 3136 PHYSICS STUDIES OF A HETEROGENEOUS LIQUID-METAL FAST BREEDER REACTOR. Lineberry, M.J.; McFarlane, H.F.; Collins, P.J.; Carpenter, S.G. (Argonne National Lab., Idaho Falls, ID). Nucl. Technol.; 44: No. 1, 21-43 (Jun 1979).

The first physics measurements for a heterogeneous design of a 1000-MW (thermal) liquid-metal fast breeder reactor were made in the Zero Power Plutonium Reactor (ZPPR) during the last half of 1976. This benchmark assembly, ZPPR-7, had a central blanket zone as well as three internal blanket rings. Fuel zones had a single enrichment. Cores with heavy plutonium buildup in the internal blankets as well as cores with clean internal blankets were investigated. Such key physics parameters as k /sub eff/, most of the important reaction rates, control rod worths, sodium void reactivity, and material worths were studied in the ZPPR-7 program. Results verified the gain in breeding that were predicted for the heterogeneous arrangement. When design-level calculations were used, calculated-to-experimental biases were different from those that had been found for homogeneous cores.

- 3137 PERFORMANCE CHARACTERISTICS OF HOMOGENEOUS VERSUS HETEROGENEOUS LIQUID-METAL FAST BREEDER REACTORS. Barthold, W.P.; Beitel, J.C. (Argonne National Lab., IL). Nucl. Technol.; 44: No. 1, 44-60 (Jun 1979).

Six homogeneous and heterogeneous core configurations with different fuel pin diameters and different degrees of neutronic coupling have been analyzed. All designs were based on the same design assumptions and constraints. Homogeneous and heterogeneous designs were evaluated with respect to fissile

inventory, flux and power distribution, burnup, breeding ratio and doubling times, sodium void reactivity, Doppler coefficient, control rod worth, and transient response. The sensitivity of breeding performance to different fuel life criteria was determined.

- 3138 PHYSICS PERFORMANCES OF A HETEROGENEOUS FAST REACTOR CORE CONCEPT STUDIED IN MASURCA. Bouget, Y.H. (Centre d'Etudes Nucleaires de Cadarache, Saint-Paul-les-Durance, France); Cosini, M.; Hammer, P.; Humbert, G.; Lyon, F.; Martini, M. Nucl. Technol.; 44: No. 1, 61-75(Jun 1979).

Experimental and theoretical work is now in progress at the Commissariat a l'Energie Atomique to support the sodium mixed-oxide fast reactor design calculations of the heterogeneous concept. The results obtained with the calculational method presently used are compared to the experimental values given by the PRE-RACINE program performed on the MASURCA facility. Critical masses, fission rate ratios, breeding gains of the different core zones, and sodium void effects are studied. The first results obtained with the new cross-section set CARNAVAL IV are given and compared to the values calculated with the previous CARNAVAL III set. The future development of the program is briefly described.

- 3139 NUCLEAR PERFORMANCE OF LIQUID-METAL FAST BREEDER REACTORS DESIGNED TO PRECLUDE ENERGETIC HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS. Bailey, H.S.; Lu, Y.S. (General Electric Co., Sunnyvale, CA). Nucl. Technol.; 44: No. 1, 76-82(Jun 1979).

A study of various core designs was conducted to investigate design features pertaining to the preclusion of energetic hypothetical core disruptive accidents. The design data and core layouts for different designs are described. A summary of the nuclear performance predicted for several typical low sodium void worth cores is presented. It appears that the tightly coupled heterogeneous core configuration offers a substantial reduction in prompt void worth and flux level at the expense of greater inventory, large flux gradients, and large changes in internal blanket power over the operating cycle.

- 3140 MULTIAXIAL CREEP EQUATIONS FOR NONLINEAR AND VOLUME-NONCONSERVATIVE IN-REACTOR CREEP. Hall, M.M. Jr. (Argonne National Lab., IL). Nucl. Technol.; 44: No. 1, 172-176(Jun 1979).

Steady-state multiaxial creep equations that may be applied to the analysis of nonlinear and volume-nonconservative in-reactor creep data have been derived. Strain rate equations are expressed in terms of the stress exponent, n , a rate constant, B , and a material parameter, β . Equivalent stress states are assumed to give equal mechanical energy dissipation rates, and the associated equivalent stress and plastic strain rate criteria are shown to be functions of both the shear and hydrostatic components of the stress and strain-rate tensors, respectively. The deviatoric creep rate coefficient is shown to be an apparent function of stress state when swelling is stress dependent and the stress exponent is greater than unity. The possible magnitude of this stress-state effect is estimated for fast-neutron-irradiated austenitic stainless steel using available microstructural models.

- 3141 CRITICAL EXPERIMENTS WITH FAST TEST REACTOR FUEL PINS IN WATER. Bierman, S.R. (Battelle-Pacific Northwest Labs., Richland,

WA); Durst, B.M.; Clayton, E.D.; Scherpelz, R.I.; Kerr, H.T. Nucl. Technol.; 44: No. 1, 141-151(Jun 1979).

A series of criticality experiments with fast test reactor (FTR) fuel pins in water has been performed in support of the Advanced Fuel Recycle Program (AFRP). The objective of these experiments was to provide clean, easily defined criticality data on AFRP-type fuel pins in water for use in verifying calculational techniques and nuclear data used in calculations. Measurement data were obtained on water-flooded square lattices of FTR fuel pins. The number of fuel pins required for criticality was determined at lattice pitches of 7.7, 9.5, 9.7, 12.6, 15.3, and 19.1 mm to be 1268, 605, 580, 279, 205, and 162, respectively. These center-to-center fuel pin spacings correspond to water-to-fuel volume ratios of 1.67, 3.33, 3.49, 6.87, 10.88, and 17.53, respectively, and cover the neutron moderation range from near optimum to the highly undermoderated. KENO-IV calculations with ENDF data from the AMPX system overestimated the experimental results by 1 to 2% in $k_{\text{sub eff}}$. KENO-IV calculations with FLANGE-ETOG-THERMOS-EGGNIIT-processed ENDF data resulted in calculated values 1 to 6% high in $k_{\text{sub eff}}$.

- 3142 U.K. DEVELOPMENT TOWARD REMOTE FABRICATION OF BREEDER REACTOR FUEL. Nelson, R.L. (UKAEA-AERA, Harwell, England); Parkinson, N.; Kent, W.C.L.; Taylor, H.A. Trans. Am. Nucl. Soc.; 32: 227-228(Jun 1979). (CONF-790602--(Summ.)).

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

REMOTE HANDLING;FABRICATION;BREEDER REACTORS; NUCLEAR FUELS;PFR REACTOR;GELS;PRECIPITATION; URANIUM DIOXIDE;PLUTONIUM DIOXIDE

- 3143 FRENCH FAST BREEDER REACTOR FUEL FABRICATION. Bailly, H.; Guillet, H.; Vialard, J.L. (CEA, Cadarache, France). Trans. Am. Nucl. Soc.; 32: 228-230(Jun 1979). (CONF-790602--(Summ.)).

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

FRANCE;SUPER PHENIX REACTOR;NUCLEAR FUELS; FABRICATION;RADIATION PROTECTION

- 3144 REMOTE FABRICATION OF FBR FUEL AT PNC. Kono, K.; Mishima, T.; Muto, T. (Power Reactor and Nuclear Fuel Development Corp., Tokai, Japan). Trans. Am. Nucl. Soc.; 32: 230-231(Jun 1979). (CONF-790602--(Summ.)).

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

PLUTONIUM;REMOTE HANDLING;FABRICATION; NUCLEAR FUELS;JOYO REACTOR;PLUTONIUM DIOXIDE; URANIUM DIOXIDE;REMOTE CONTROL;MONJU REACTOR; FUEL FABRICATION PLANTS

- 3145 REMOTE REPROCESSING OF METAL FUEL FOR THE EBR-II. Dunworth, R.J. (Argonne National Lab., Idaho Falls, ID). Trans. Am. Nucl. Soc.; 32: 278(Jun 1979). (CONF-790602--(Summ.)).

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

EBR-2 REACTOR;FUEL REPROCESSING PLANTS;FUEL CYCLE;SPENT FUELS;REPROCESSING

- 3146 INTERLABORATORY COMPUTATIONAL COMPARISONS OF CRITICAL FAST TEST REACTOR PIN LATTICES. Mincey, J.F. (Oak Ridge National Lab., TN); Kerr, H.T.; Durst, B.M. Trans. Am. Nucl. Soc.; 32: 323-325(Jun 1979). (CONF-

790602--(Summ.)).

From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).

SPENT FUELS;REPROCESSING;CRITICALITY;LMFBR
TYPE REACTORS;FUEL REPROCESSING PLANTS;
COMPARATIVE EVALUATIONS;COMPUTERIZED SIMULATION;
FUEL PINS

- 3147 DESIGN OF ELEVATED TEMPERATURE PIPING
FOR ADVANCED NUCLEAR PLANTS. Chang, T.Y.;
Kent, G.F.; Kuo, C.S. (Stone and Webster Eng
Corp, Boston, Mass). Am. Soc. Mech. Eng.,
[Pap.]; No. 79-NE-7, 1-12(Jun 1979).

This paper reviews the experience in design
and analysis accumulated by Stone and Webster
Engineering Corporation on elevated temperature
piping for advanced nuclear power generation
facilities. The presentation consists of two
parts: Part I describes existing and proposed
design criteria and procedures for elevated
temperature ASME Code Class 1, 2 and 3 piping.
Part II demonstrated the application of these
criteria and procedures to elevated temperature
nuclear piping systems of a High Temperature
Gas Cooled Reactor (HTGR) and a Prototype Large
Breeder Reactor (PLBR) plant.

- 3148 IHX DESIGN FOR POOL TYPE LMFBR SYSTEM
APPLICATION. DeFur, D.D. (Combust Eng, Inc,
Chattanooga, Tenn). Am. Soc. Mech. Eng.,
[Pap.]; No. 79-NE-3, 1-5(Jun 1979).

This paper presents the design of a 500 MWt
Intermediate Heat Exchanger (IHX) for
application in a multi-zone pool-type Liquid
Metal Fast Breeder Reactor System. The primary
feature around which this design was developed
is the ability to function under all steady-
state and transient conditions without the need
for a distinct mechanical feature to
accommodate differential thermal expansion
between the tube bundle and shell. This is
accomplished by using thermal baffles and
judiciously adjusting the ratio of shell length
in contact with the hot and cold pools of the
multi-zone reactor. Thus, the component shell
is maintained at a higher temperature than the
tube bundle and the tubes are basically tension
members. Under certain transient conditions the
tubes heat up and become compression members
but are not loaded to the point of buckling. A
secondary, but not less significant, feature is
the low primary side pressure drop and good
flow distribution offered by this design.

- 3149 HIGH RELIABILITY STRAIGHT TUBE LMFBR
STEAM GENERATOR DESIGN. Spring, A.H.; DeFur,
D.D. (Combust Eng, Inc, Chattanooga, Tenn).
Am. Soc. Mech. Eng., [Pap.]; No. 79-NE-4, 1-
5(Jun 1979).

Straight-tube counterflow steam generators
for LMFBR applications have been proposed,
tested, fabricated and operated with varying
degrees of success. A new embodiment of the
straight-tube concept is described which
incorporates a number of unique features which
contribute to high reliability and
availability. These features include a
replaceable bellows for accommodation of
differential thermal expansion between shell
and tubes and a redundant, crevice-free tube-to-
tubesheet joint design. The design can also
easily incorporate single-wall or double-wall
tubes. Single and double-wall tube versions are
described whose thermal and geometric size are
based on anticipated manufacturing limitations.
The results of scoping tests of the tube-to-
tubesheet welds are described which provide
positive indications of the soundness of the
weld design.

- 3150 PRELIMINARY ANALYSIS AND SCREENING
CRITERIA FOR ELEVATED TEMPERATURE PIPING.
Minami, H.M.; Naugle, F.V. (Rockwell Int,
Canoga Park, Calif). Am. Soc. Mech. Eng.,
[Pap.]; No. 79-NE-6, 1-7(Jun 1979).

The complexities of the ASME Class 1 nuclear
elevated temperature structural design criteria
precludes the possibility of performing a
complete Code Case 1592 analysis for all of the
alternative designs normally investigated in
the course of the design of a Liquid Metal Fast
Breeder Reactor (LMFBR) high-temperature piping
system. To enable the piping design to proceed
on an orderly controlled basis, the piping
design analysis is separated into the following
phases: screening analysis to determine pipe
routing feasibility, preliminary analysis to
determine pipe support configuration, final
design analysis (including transients) to
demonstrate Code compliance. The application of
the first phases of this procedure is
demonstrated by a summary of the analyses
performed in the development of the preliminary
design configuration for the primary system of
a commercial-sized LMFBR.

- 3151 IMPACT OF BUILDING DESIGN ON AUXILIARY
LIQUID METAL PIPING. Carter, A.L.; McNamara,
C.P. (Holmes and Narver, Inc, Orange, Calif).
Am. Soc. Mech. Eng., [Pap.]; No. 79-NE-8, 1-
5(Jun 1979).

The routing of elevated temperature liquid
metal piping within the constraints imposed by
a building design which becomes fixed during
the early design stages presents a set of
unique design problems. In addition to the
building size and layout, other factors such as
code requirements, system functions, inspection
and maintenance must be considered. This paper
examines the approach and methodology utilized
to arrive at an optimum routing for the
auxiliary liquid metal piping in a Liquid Metal
Fast Breeder Reactor (LMFBR) Plant during the
early design stages.

- 3152 SENSITIVITY ANALYSIS OF INTEGRAL
PARAMETERS TO SELECTED CROSS SECTIONS FOR AN
LMFBR HETEROGENEOUS CORE. Bruna, G. (NIRA,
Genova, Italy); Conversano, R.; D'Angelo, A.;
Gastaldo, G.; Oliva, A.; Rinaldis, D. Trans.
Am. Nucl. Soc.; 32: 763-766(Jun 1979). (CONF-
790602--(Summ.)).

From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).

LMFBR TYPE REACTORS;REACTOR KINETICS;REACTOR
CORES;OPTIMIZATION;MATHEMATICAL MODELS

- 3153 TECHNIQUE FOR SENSITIVITY ANALYSIS OF
SPACE- AND ENERGY-DEPENDENT BURNUP
CALCULATIONS. Williams, M.L. (Oak Ridge
National Lab., TN); White, J.R.; Burns, T.J.
Trans. Am. Nucl. Soc.; 32: 766-768(Jun 1979).
(CONF-790602--(Summ.)).

From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).

LMFBR TYPE REACTORS;REACTOR KINETICS;BURNUP;
NEUTRON FLUX;MATHEMATICAL MODELS

- 3154 ANALYSIS OF EXPOSURE-DEPENDENT
HETEROGENEOUS FBR SPECTRA USING CONTINUOUS
SLOWING DOWN INTEGRAL TRANSPORT CONCEPTS.
Badruzzaman, A.; Badruzzaman, T.; Becker, M.
(Rensselaer Polytechnic Inst., Troy, NY).
Trans. Am. Nucl. Soc.; 32: 768-769(Jun 1979).
(CONF-790602--(Summ.)).

From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).

LMFBR TYPE REACTORS;REACTOR KINETICS;NEUTRON
SPECTRA;NEUTRON TRANSPORT THEORY

- 3155 EVALUATION OF STOCHASTIC VOID MODELS FOR BUBBLE WORTH CALCULATIONS. Vaughan, E.U. (Rockwell International, Canoga Park, CA); Hoffman, T.J. Trans. Am. Nucl. Soc.; 32: 772-773(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS; REACTOR CORES; BUBBLES; REACTOR KINETICS; REACTIVITY WORTHS; VOID FRACTION
- 3156 MEASUREMENT OF THE DOPPLER REACTIVITY WORTH OF UO_2 AND ThO_2 IN AN LMFBR SPECTRUM. Bhattacharyya, S.K. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 32: 784-785(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS; REACTOR KINETICS; URANIUM DIOXIDE; THORIUM OXIDES; DOPPLER COEFFICIENT; REACTIVITY WORTHS
- 3157 LOADING ERROR SIMULATIONS AND CONTROL WORTH MEASUREMENTS IN A ZPPR CRITICAL ASSEMBLY OF HETEROGENEOUS CONFIGURATION. Lu, Y.S.; Crick, S.C.; Stewart, S.L. (General Electric Corp., Sunnyvale, CA). Trans. Am. Nucl. Soc.; 32: 785-787(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS; REACTOR FUELING; REACTOR ACCIDENTS; REACTIVITY WORTHS; SIMULATION; ZPR REACTOR; CONTROL ROD WORTHS
- 3158 THEORETICAL EVALUATION OF EX-VESSEL MONITORING FOR INITIAL FUEL LOADING OF A LIQUID-METAL FAST BREEDER REACTOR. Mihalcz, J.T.; Selby, D.L.; Klee, H.E. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 32: 787-788(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
CLINCH RIVER BREEDER REACTOR; REACTOR FUELING; REACTOR KINETICS; CRITICALITY; REACTOR INSTRUMENTATION; PERFORMANCE
- 3159 NUCLEAR UNCERTAINTIES AS A CONSIDERATION IN REACTOR DESIGNS. Kujawski, E. (General Electric Corp., Sunnyvale, CA); Kallfelz, J.M.; Marable, J.H. Trans. Am. Nucl. Soc.; 32: 791-792(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS; REACTOR KINETICS; FUEL CYCLE; COMPUTER CALCULATIONS; NUCLEAR DATA COLLECTIONS
- 3160 LMFBR THERMAL HYDRAULICS. Fauske, H.K. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 32: 801(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS; REACTOR CORES; PRIMARY COOLANT CIRCUITS; HYDRAULICS; HEAT TRANSFER; TWO-PHASE FLOW
- 3161 THERMAL MIXER MIXING CORRELATIONS FOR BOTH SODIUM AND WATER. Kasza, K.E. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 32: 825-826(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS; PIPES; PIPE FITTINGS; MIXING; HYDRAULICS; TEMPERATURE
- GRADIENTS
- 3162 COBRA-IV COMPARISON WITH THE W-ARD 11:1 LMFBR AIR FLOW TEST. Basehore, K.L.; George, T.L. (Battelle Northwest Labs., Richland, WA). Trans. Am. Nucl. Soc.; 32: 826-827(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS; FUEL ASSEMBLIES; REACTOR CORES; HYDRAULICS; HEAT TRANSFER; COMPUTER CALCULATIONS
- 3163 FLOW INSTABILITY ANALYSIS IN SODIUM-HEATED ONCE-THROUGH STEAM GENERATORS. Suzuoki, A.; Yamakawa, M.; Sagawa, N. (Hitachi Energy Research Lab., Japan). Trans. Am. Nucl. Soc.; 32: 831-832(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS; STEAM GENERATORS; HYDRAULICS; LIQUID FLOW
- 3164 INFLUENCE OF ASYMMETRIC PHENOMENA ON FUEL PIN BREACH. Patel, M.R.; Harbourn, B.L. (General Electric Corp., Sunnyvale, CA). Trans. Am. Nucl. Soc.; 32: 216-217(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS; FUEL ELEMENT FAILURE; STRESS ANALYSIS; STRAINS; COMPUTER CALCULATIONS
- 3165 COMPARISON OF IN-REACTOR AND OUT-OF-REACTOR FUEL PIN CLADDING STRAIN UNDER TRANSIENT LOADING. Duncan, D.R.; Hunter, C.W.; Thielges, J.R.; Hikido, T.; Hanson, J.E. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 32: 221-223(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS; TREAT REACTOR; TRANSIENTS; FUEL PINS; STRAINS; FUEL CANS; TESTING
- 3166 CLADDING DAMAGE ANALYSIS APPLIED TO CARBIDE TREAT TESTS HC2 AND HC3. Tsai, H.C.; Neimark, L.A. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 32: 223-224(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
FUEL PINS; FUEL CANS; FFTF REACTOR; TRANSIENTS; STRAINS; TREAT REACTOR
- 3167 DEVELOPMENT OF ANNEALED ALLOY 800 FOR ELEVATED-TEMPERATURE NUCLEAR APPLICATIONS. Sessions, C.E.; Langford, P.J.; Duke, J.M. (Westinghouse Electric Corp., Tampa, FL). Trans. Am. Nucl. Soc.; 32: 268(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS; STEAM GENERATORS; INCOLOY 800; CREEP; STRESSES; ANNEALING; TENSILE PROPERTIES
- 3168 LONG-TERM IN-REACTOR CORROSION OF STAINLESS-STEEL CLADDING IN LIQUID SODIUM. Jensen, B.W.; Weber, J.W.; Jost, J.W. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 32: 265-266(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS; STAINLESS STEEL-316; FUEL CANS; CORROSION; SODIUM; CORROSIVE EFFECTS

- 3169 STRESS-RUPTURE PROPERTIES OF SODIUM-DECARBURIZED AND THERMALLY AGED 2 1/4 CR-1 MO STEEL. Klueh, R.L. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 32: 262-263(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;STEAM GENERATORS;CHROMIUM-MOLYBDENUM STEELS;DECARBURIZATION;STRESSES;RUPTURES;AGING;HIGH TEMPERATURE;MECHANICAL PROPERTIES
- 3170 EFFECTS OF STOICHIOMETRY ON CLADDING REACTION IN MIXED OXIDE FUEL. Lawrence, L.A. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 32: 249-250(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;FUEL-CLADDING INTERACTIONS;STOICHIOMETRY;URANIUM DIOXIDE;PLUTONIUM DIOXIDE;STAINLESS STEEL-316;BURNUP;FUEL ELEMENTS
- 3171 INFLUENCE OF OXYGEN-TO-METAL RATIO ON MIXED OXIDE FUEL PERFORMANCE. Lawrence, L.A.; Leggett, R.D. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 32: 247-248(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;URANIUM DIOXIDE;PLUTONIUM DIOXIDE;STOICHIOMETRY;PERFORMANCE;FUEL-CLADDING INTERACTIONS;FISSION PRODUCTS;SOLID FUELS
- 3172 SODIUM COMPONENT RELIABILITY DATA COLLECTION AT CREDO. Bott, T.F.; Haas, P.M.; Manning, J.J. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 32: 480-482(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS;REACTOR COMPONENTS;RELIABILITY;DATA COMPILATION;DATA ANALYSIS
- 3173 SERP DEPOSITION FROM TESTING AND CLEANING AN LMFBR GENERATOR. Neely, H.H.; Stearns, J.D.; Shepard, R.C.; McCoilum, W.A. (Rockwell International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 32: 692-693(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;STEAM GENERATORS;TEST FACILITIES;MOCKUP;LEAKS;MOLTEN METAL-WATER REACTIONS;DEPOSITS
- 3174 SIMPLE PROCEDURE FOR DETERMINING IMPLICATIONS OF DESIGN CHANGES ON FAST REACTOR FUEL CYCLE COST AND BREEDING PERFORMANCE. Badruzzaman, A.; Becker, M. (Rensselaer Polytechnic Inst., Troy, NY). Trans. Am. Nucl. Soc.; 32: 405-406(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;FUEL CYCLE;COST;PLANNING
- 3175 COMPUTER MODELING OF THE DYNAMIC BEHAVIOR OF DELAYED-NEUTRON CONCENTRATIONS IN EBR-II. Gross, K.C.; Strain, R.V. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 32: 798-799(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
- (3 Jun 1979).
EBR-2 REACTOR;REACTOR INSTRUMENTATION;FAILED ELEMENT MONITORS;COMPUTER CALCULATIONS;MATHEMATICAL MODELS;ACCURACY
- 3176 OBSERVATIONS OF IN-REACTOR ENDURANCE AND RUPTURE LIFE FOR FUELED AND UNFUELED FTR CLADDING. Lovell, A.J.; Christensen, B.V.; Chin, B.A. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 32: 217-218(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
FFTF REACTOR;FUEL ELEMENT FAILURE;FUEL CANS;RUPTURES;STAINLESS STEEL-316;PERFORMANCE TESTING
- 3177 OBSERVATIONS OF IN-REACTOR STRAIN FOR FUELED AND UNFUELED FTR CLADDING. Gilbert, E.R.; Makenas, B.J.; Wilson, D.R. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 32: 218-219(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
FFTF REACTOR;FUEL CANS;STRAINS;CREEP;SWELLING
- 3178 CUMULATIVE DAMAGE AND WEIBULL ANALYSES OF EBR-II DRIVER FUEL LIFETIME. Seidel, B.R.; Shields, J.A. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 32: 219-221(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
EBR-2 REACTOR;FUEL ELEMENTS;SERVICE LIFE
- 3179 PERSPECTIVE ON THE EBR-II REMOTE FABRICATION EXPERIENCE. Feldman, M.J. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 32: 225-226(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
EBR-2 REACTOR;FUEL ELEMENTS;FABRICATION;SOLID FUELS;REMOTE HANDLING
- 3180 EFFECT OF A TIME-VARYING POWER LEVEL IN EBR-II ON MIXED OXIDE FUEL BURNUP. Stone, I.Z.; Jost, J.W.; Baker, R.B. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 32: 282-283(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
EBR-2 REACTOR;BURNUP;POWER DENSITY;URANIUM DIOXIDE;PLUTONIUM DIOXIDE
- 3181 POSSIBLE METAL AND OXIDE FUEL LOADINGS OF EBR-II UTILIZING ²³²Th/²³³U. Foltman, A.J. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 32: 283-284(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
EBR-2 REACTOR;SOLID FUELS;THORIUM CYCLE
- 3182 NEUTRON SOURCE STRENGTH IN FFTF FUEL AS A FUNCTION OF IRRADIATION. Ombrellaro, P.A.; Petrowicz, N.E. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 32: 736-737(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
FFTF REACTOR;REACTOR CORES;FUEL ASSEMBLIES;REACTIVITY WORTHS;BURNUP

- 3183 ANALYSIS OF A LOSS-OF-PIPING-INTEGRITY TRANSIENT IN AN LMFBR FUEL ASSEMBLY. Domanus, H.M.; Chen, M.J.; Sha, W.T. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 32: 827-829(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;FUEL ASSEMBLIES;LOSS OF COOLANT;REACTOR SAFETY;HYDRAULICS;HEAT TRANSFER; COMPUTER CALCULATIONS
- 3184 BUOYANCY EFFECTS ON PRESSURE LOSS IN ROD ARRAYS UNDER LAMINAR FLOW. Yang, J.W. (Brookhaven National Lab., Upton, NY). Trans. Am. Nucl. Soc.; 32: 829-830(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;REACTOR SAFETY; LOSS OF FLOW;NATURAL CONVECTION;REACTOR CORES; FUEL ASSEMBLIES;HYDRAULICS;HEAT TRANSFER
- 3185 SOLUTION OF RADIATION-CONDUCTION PROBLEMS IN LMFBR SAFETY WITH COLLOCATION METHOD. Chawla, T.C. (Argonne National Lab., IL); Chan, S.H.; Hauser, G.M. Trans. Am. Nucl. Soc.; 32: 287(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;REACTOR CORE DISRUPTION; MELTDOWN;RADIANT HEAT TRANSFER;NUMERICAL ANALYSIS;SERIES EXPANSION
- 3186 REGIONAL-TAILORING OF EBR-II FOR SLOW TRANSIENT TESTS. Grimm, K.N.; Franklin, F.C.; Meneghetti, D. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 32: 501-502(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;REACTOR SAFETY; TRANSIENTS;FUEL PINS;DAMAGE;THERMAL STRESSES; TEMPERATURE GRADIENTS;EBR-2 REACTOR
- 3187 EFFECT OF PUMP-STORED KINETIC ENERGY ON NATURAL-CIRCULATION DECAY-HEAT REMOVAL IN LOOP-TYPE LMFBRs. Madni, I.K.; Khatib-Rahbar, M.; Agrawal, A.K. (Brookhaven National Lab., Upton, NY). Trans. Am. Nucl. Soc.; 32: 502-503(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;REACTOR SAFETY;LOSS OF FLOW;RHR SYSTEMS;AFTER-HEAT;AFTER-HEAT REMOVAL; NATURAL CONVECTION;HEAT TRANSFER;TEMPERATURE GRADIENTS;PERFORMANCE
- 3188 TRANSITION TO NATURAL CONVECTION FROM LOW-TEMPERATURE, LOW-FLOW CONDITIONS IN LOOP-TYPE LMFBRs. Khatib-Rahbar, M.; Madni, I.K.; Agrawal, A.K.; Cazzoli, E.G. (Brookhaven National Lab., Upton, NY). Trans. Am. Nucl. Soc.; 32: 503-504(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;REACTOR SAFETY;LOSS OF FLOW;AFTER-HEAT REMOVAL;PERFORMANCE;HEAT TRANSFER;NATURAL CONVECTION;HYDRAULICS
- 3189 FIRE HAZARDS ANALYSIS OF BARRIERS: AN ANALYTICAL APPROACH. Berry, D.L. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 32: 504-505(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;FIRE PREVENTION; ENGINEERED SAFETY SYSTEMS;PERFORMANCE
- 3190 AERODYNAMIC PROPERTIES OF AEROSOLS FORMED BY A SODIUM POOL FIRE. Patel, S.; Wegrzyn, J.; Shaw, D.T. (State Univ. of New York, Buffalo). Trans. Am. Nucl. Soc.; 32: 505-506(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;REACTOR SAFETY; REACTOR CORE DISRUPTION;FIRES;SODIUM;AEROSOLS; AERODYNAMICS
- 3191 COMPARISON OF THE HAARM-3 FALLOUT MODEL WITH NUCLEAR SAFETY PILOT PLANT (NSPP) DATA. Han, J.T.; Kress, T.S.; Adams, R.E. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 32: 506-508(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;REACTOR SAFETY; REACTOR CORE DISRUPTION;AEROSOLS;SODIUM OXIDES; CONTAINMENT SYSTEMS;COMPUTER CALCULATIONS
- 3192 RELAP4/SNR: A PROGRAM TO SIMULATE STEAM GENERATOR TRANSIENTS OF FBRs. Schuetzle, R. (Univ. of Stuttgart, Germany); Friedel, T.; Wehner, H. Trans. Am. Nucl. Soc.; 32: 508-509(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
SNR-1 REACTOR;REACTOR SAFETY;STEAM GENERATORS;TRANSIENTS;HYDRAULICS;HEAT TRANSFER; TEMPERATURE GRADIENTS;COMPUTER CALCULATIONS
- 3193 RESPONSE OF CRACKED HEXAGONAL SUBASSEMBLY DUCTS TO IMPACT LOADING. Glazik, J.L. Jr.; Petroski, H.J. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 32: 511-512(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;FUEL ASSEMBLIES; FUEL CHANNELS;CRACKS;REACTOR SAFETY;IMPACT SHOCK;IMPACT STRENGTH
- 3194 NEW EULERIAN HYDRODYNAMIC TECHNIQUE FOR ANALYZING FBR CONTAINMENT WITH COMPLEX INTERNAL STRUCTURES. Wang, C.Y.; Chang, Y.W. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 32: 512-513(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;CONTAINMENT SHELLS; REACTOR CORE DISRUPTION;HYDRODYNAMICS;STRESS ANALYSIS;PRESSURE GRADIENTS;MATHEMATICAL MODELS
- 3195 STRUCTURAL RESPONSE OF THE UPPER INTERNALS. Kennedy, J.M. (Argonne National Lab., IL); Belytschko, T.B. Trans. Am. Nucl. Soc.; 32: 513-514(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).
LMFBR TYPE REACTORS;REACTOR SAFETY; REACTOR CORE DISRUPTION;REACTOR INTERNALS; HYDRODYNAMICS;STRESS ANALYSIS;DEFORMATION
- 3196 COMPARISON OF ALICE PREDICTIONS WITH SRI FLEXIBLE VESSEL EXPERIMENTS. Chu, H.Y.; Chang, Y.W. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 32: 515-516(Jun 1979). (CONF-790602--(Summ.)).

- From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS;CONTAINMENT
SHELLS;STRESS ANALYSIS;REACTOR SAFETY;REACTOR
CORE DISRUPTION;HYDRODYNAMICS;COMPUTER
CALCULATIONS
- 3197 COMPARISON OF ICECO CODE PREDICTIONS
WITH COVA EXPERIMENTAL DATA. Wang, C.Y.;
Chang, Y.W. (Argonne National Lab., IL).
Trans. Am. Nucl. Soc.; 32: 516-517(Jun 1979).
(CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS;REACTOR SAFETY;
REACTOR CORE DISRUPTION;CONTAINMENT SYSTEMS;
STRESS ANALYSIS;PRESSURE GRADIENTS;COMPUTER
CALCULATIONS
- 3198 MILD TRANSIENT OVERPOWER ACCIDENTS WITH
ADVANCED FUELS. Henry, T.P.; Siegmann, E.R.
(Combustion Engineering, Windsor, CT). Trans.
Am. Nucl. Soc.; 32: 518-519(Jun 1979). (CONF-
790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS;REACTOR SAFETY;
TRANSIENT OVERPOWER ACCIDENTS;URANIUM CARBIDES;
PLUTONIUM CARBIDES;FUEL PINS;FUEL ELEMENT
FAILURE;PERFORMANCE TESTING
- 3199 STUDY OF CORE DISRUPTIVE ACCIDENTS IN A
LARGE METAL-FUELED LMFBR. Bowers, C.H.;
Briggs, L.L.; Ferguson, D.R. (Argonne
National Lab., IL). Trans. Am. Nucl. Soc.; 32:
519-520(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR TYPE REACTORS;REACTOR SAFETY;REACTOR
CORE DISRUPTION;REACTOR CORES;THERMAL STRESSES;
TEMPERATURE GRADIENTS;PRESSURE GRADIENTS
- 3200 CLADDING MOTION IN A 37-PIN LOSS-OF-FLOW
EXERIMENT: SLSF P3. Kuzay, T.M.; Bordner,
G.L.; Kraft, T.E.; Marr, W.W. (Argonne
National Lab., IL). Trans. Am. Nucl. Soc.; 32:
520-522(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS;REACTOR SAFETY;
FUEL PINS;LOSS OF FLOW;MELTDOWN;FLOW BLOCKAGE;
THERMAL STRESSES;SIMULATION;FUEL ELEMENT
FAILURE
- 3201 PHENOMENOLOGICAL MODELING OF SRI
CONDENSIBLE VAPOR SOURCE EXPERIMENTS.
Corradini, M.L.; Rohsenow, W.M.; Todreas, N.E.
(Massachusetts Inst. of Tech., Cambridge).
Trans. Am. Nucl. Soc.; 32: 522-523(Jun 1979).
(CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS;REACTOR SAFETY;
REACTOR CORE DISRUPTION;HEAT TRANSFER;
HYDRAULICS;VAPORS
- 3202 MICROWAVE HEATING SIMULATIONS OF FISSION
ENERGY GENERATION IN VOLUME-BOILING POOL
SYSTEMS. Makowitz, H.; Ginsberg, T.
(Brookhaven National Lab., Upton, NY). Trans.
Am. Nucl. Soc.; 32: 523-524(Jun 1979). (CONF-
790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS;REACTOR SAFETY;
LOSS OF FLOW;MELTDOWN;HYDRAULICS;HEAT TRANSFER;
CORIUM
- 3203 FILM BOILING AND VAPOR EXPLOSIONS FROM
SMALL SPHERES. Shih, C.; El-Wakil, M.M.
(Univ. of Wisconsin, Madison). Trans. Am.
Nucl. Soc.; 32: 524-525(Jun 1979). (CONF-
790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS;REACTOR SAFETY;
REACTOR ACCIDENTS;FILM BOILING;HEAT TRANSFER
- 3204 MOLTEN CORE INTERACTIONS WITH COMMERCIAL
FIREBRICKS. Swanson, D.G. (Aerospace Corp.,
Los Angeles, CA); van Paassen, H.L.L.;
Marchese, A.R. Trans. Am. Nucl. Soc.; 32:
525-526(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS;REACTOR SAFETY;
MELTDOWN;CORE CATCHERS;PERFORMANCE TESTING;
ALUMINIUM OXIDES;SILICON OXIDES;URANIUM DIOXIDE;
REACTOR MATERIALS;CORIUM
- 3205 FORMATION OF BUBBLE BOUNDARY LAYER OVER
AN INCLINED POROUS PLATE. Cheung, F.B.;
Farhadieh, R. (Argonne National Lab., IL).
Trans. Am. Nucl. Soc.; 32: 526-527(Jun 1979).
(CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS;REACTOR SAFETY;
MELTDOWN;CONTAINMENT SHELLS;CONCRETES;CORIUM;
HEAT TRANSFER;SIMULATION;REACTOR MATERIALS
- 3206 EXPERIMENTAL STUDY OF THE DOWNWARD
PENETRATION OF A HOT LIQUID POOL INTO THE
HORIZONTAL SURFACE OF A GAS-RELEASING SOLID.
Farhadieh, R. (Argonne National Lab., IL);
Peerenboom, J. Trans. Am. Nucl. Soc.; 32:
527-528(Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR. LMFBR TYPE REACTORS;REACTOR SAFETY;
MELTDOWN;AFTER-HEAT;AFTER-HEAT REMOVAL;CORIUM;
HYDRAULICS
- 3207 OXYGEN-TO-METAL RATIO CONTROL DURING
FABRICATION OF MIXED OXIDE FAST BREEDER REACTOR
FUEL PELLETS. Rasmussen, D.E.; Benecke, M.W.;
Jentzen, W.R.; McCord, R.B. (Westinghouse
Electric Corp., Richland, WA). Trans. Am.
Nucl. Soc.; 32: 246(Jun 1979). (CONF-790602--
(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
PLUTONIUM DIOXIDE;URANIUM DIOXIDE;FUEL
PELLETS;FABRICATION;STOICHIOMETRY;LMFBR TYPE
REACTORS
- 3208 LMFBR STEAM CYCLE SELECTION. McDonald,
J.S. (Rockwell International, Canoga Park,
CA). Trans. Am. Nucl. Soc.; 32: 558-559(Jun
1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR TYPE REACTORS;STEAM SYSTEMS;
SUPERHEATING;COMPARATIVE EVALUATIONS;
PERFORMANCE
- 3209 CONSIDERATIONS IN SELECTION OF STEAM
CONDITIONS FOR SODIUM-COOLED FAST BREEDER
REACTORS. Holmes, J.A.G. (Nuclear Power
Company, Ltd., Warrington, England). Trans.
Am. Nucl. Soc.; 32: 559-560(Jun 1979). (CONF-
790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR TYPE REACTORS;STEAM SYSTEMS;DESIGN;

- CHROMIUM-MOLYBDENUM STEELS; CORROSION RESISTANCE;
STEAM GENERATORS
- 3210 COMPARISON OF LMFBR STEAM CYCLES.
Anderson, C.A. Jr. (Westinghouse Advanced
Reactors Division, Madison, PA). Trans. Am.
Nucl. Soc.; 32: 560-561 (Jun 1979). (CONF-
790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
PLBR REACTOR; STEAM SYSTEMS; COMPARATIVE
EVALUATIONS; STEAM GENERATORS; DESIGN
- 3211 COMPARISON OF LMFBR STEAM CYCLE
CONDITIONS CONSIDERING FUTURE TECHNICAL
IMPROVEMENTS. Aubert, H. (CEA, Cadarache,
France); Baer, A.; Hirs, G.; Jansing, W.;
Koehler, M. Trans. Am. Nucl. Soc.; 32: 561-
562 (Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR TYPE REACTORS; STEAM SYSTEMS; THERMAL
EFFICIENCY; COMPARATIVE EVALUATIONS
- 3212 CHECKLIST FOR COMPARING LMFBR STEAM
CYCLES. Davies, S.M. (General Electric Co.,
Sunnyvale, CA); Lutz, D.E.; Wyckoff, H.L.
Trans. Am. Nucl. Soc.; 32: 562-564 (Jun 1979).
(CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR TYPE REACTORS; STEAM SYSTEMS;
COMPARATIVE EVALUATIONS; THERMODYNAMICS
- 3213 LMFBR STEAM CYCLES: IS EFFICIENCY THE
ULTIMATE GOAL. Kendall, J.M. (Electric
Power Research Inst., Palo Alto, CA). Trans.
Am. Nucl. Soc.; 32: 564-565 (Jun 1979). (CONF-
790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR TYPE REACTORS; STEAM SYSTEMS; THERMAL
EFFICIENCY; PLBR REACTOR
- 3214 IMPACT OF FUEL CYCLE COST OPTIMIZATION
ON THE DESIGN OF METAL-FUELED LMFBRs. Lam,
P.S.K. (Argonne National Lab., IL). Trans.
Am. Nucl. Soc.; 32: 570-571 (Jun 1979). (CONF-
790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR TYPE REACTORS; REACTOR CORES; DESIGN;
FUEL ELEMENTS; FUEL CYCLE; COST; OPTIMIZATION
- 3215 HETEROGENEOUS OXIDE REFERENCE,
TRANSMUTER, DENATURED CORE DESIGNS. Paik,
N.C.; Doncals, R.A.; Lake, J.A.; Rathbun, R.W.;
Varner, S.K.; Petras, D.S.; Yarbrough, M.B.
(Westinghouse Advanced Reactors Division,
Madison, PA). Trans. Am. Nucl. Soc.; 32: 571-
574 (Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
Proliferation-resistant large core design
study. LMFBR TYPE REACTORS; FUEL CYCLE;
COMPARATIVE EVALUATIONS; DENATURED FUEL; ACTINIDE
BURNER REACTORS; REACTOR CORES; DESIGN
- 3216 ANALYTICAL METHOD TO ACCURATELY PREDICT
LMFBR CORE FLOW DISTRIBUTION. Carelli, M.D.;
Willis, J.M. (Westinghouse Advanced Reactors
Division, Madison, PA). Trans. Am. Nucl. Soc.;
32: 575-576 (Jun 1979). (CONF-790602--
(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
CATFISH code. LMFBR TYPE REACTORS; REACTOR
CORES; FLUID FLOW; HYDRAULICS; MATHEMATICAL MODELS;
COMPUTER CODES; C CODES
- 3217 DEVELOPMENTS IN ELECTROLYTIC OXYGEN
METER PROBES FOR SODIUM USE. Barton, G.B.
(Westinghouse Electric Corp., Richland, WA).
Trans. Am. Nucl. Soc.; 32: 621-622 (Jun 1979).
(CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR TYPE REACTORS; REACTOR COOLING SYSTEMS;
OXYGEN METERS; PERFORMANCE TESTING; SODIUM; PROBES
- 3218 BLANKET ASSEMBLY FLOW REGULATING DEVICE.
Howard, P.A.; Carlson, R.D. (Argonne National
Lab., IL). Trans. Am. Nucl. Soc.; 32:
627 (Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
BREEDING BLANKETS; LMFBR TYPE REACTORS; FLUID
FLOW; FLOW REGULATORS; ORIFICES; TEMPERATURE
DEPENDENCE; DESIGN; PERFORMANCE TESTING
- 3219 FBR CORE TRANSIENT SIMULATION. Guppy,
J.G.; Agrawal, A.K. (Brookhaven National
Lab., Upton, NY). Trans. Am. Nucl. Soc.; 32:
529-530 (Jun 1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR TYPE REACTORS; TRANSIENTS; HYDRAULICS;
HEAT TRANSFER; MATHEMATICAL MODELS; COMPUTER
CALCULATIONS; B CODES; S CODES
- 3220 WHOLE-CORE THERMAL-HYDRAULIC TRANSIENT
CODE DEVELOPMENT AND VERIFICATION FOR LMFBR
ANALYSIS. Spencer, D.R. (Westinghouse
Advanced Reactors Division, Madison, PA).
Trans. Am. Nucl. Soc.; 32: 530-531 (Jun 1979).
(CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR TYPE REACTORS; TRANSIENTS; REACTOR CORES;
HYDRAULICS; COMPUTER CODES; HEAT TRANSFER
- 3221 MODIFIED VERSION OF COBRA FOR WHOLE-CORE
LMFBR TRANSIENT ANALYSIS. George, T.L.;
Masterson, R.E.; Basehore, K.L. (Battelle
Northwest Labs., Richland, WA). Trans. Am.
Nucl. Soc.; 32: 531-532 (Jun 1979). (CONF-
790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
LMFBR TYPE REACTORS; REACTOR CORES; TRANSIENTS;
HEAT TRANSFER; TEMPERATURE DISTRIBUTION; COMPUTER
CODES; C CODES
- 3222 CORA: TRANSIENT ANALYSIS CODE FOR A
CLUSTER OF REACTOR CORE ASSEMBLIES. Johnson,
H.G. (Westinghouse Electric Corp., Richland,
WA). Trans. Am. Nucl. Soc.; 32: 533-534 (Jun
1979). (CONF-790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
FFTF REACTOR; TRANSIENTS; HYDRAULICS; HEAT
TRANSFER; FUEL ASSEMBLIES; COMPUTER CODES; C CODES
- 3223 THERMAL-HYDRAULIC MODELING FOR THE
ANALYSIS OF LMFBR CORE RESTRAINT TRANSIENT
RESPONSES. Fox, J.N.; Wei, J.P. (General
Electric Co., Sunnyvale, CA). Trans. Am.
Nucl. Soc.; 32: 534-535 (Jun 1979). (CONF-
790602--(Summ.)).
From ANS annual meeting; Atlanta, GA, USA
(3 Jun 1979).
CORTAC-30 code. LMFBR TYPE REACTORS; REACTOR
CORE RESTRAINTS; HYDRAULICS; HEAT TRANSFER;
TRANSIENTS; COMPUTER CODES; C CODES

3224 TRANSIENT CODE FOR ANALYZING LMFBR SUBASSEMBLIES INCLUDING INTERSUBASSEMBLY HEAT TRANSFER (TCLUST1). Chang, L.K.; Feldman, E.E. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 32: 535-537(Jun 1979). (CONF-790602--(Summ.)).

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

LMFBR TYPE REACTORS;HEAT TRANSFER;FUEL ELEMENT CLUSTERS;TRANSIENTS;HYDRAULICS;COMPUTER CODES;T CODES

3225 CORTRAN CODE FOR WHOLE-CORE LMFBR TRANSIENT THERMAL-HYDRAULIC ANALYSIS. Khan, E.U. (Battelle Northwest Labs., Richland, WA); Hinds, A.; Cheatham, R.L. Trans. Am. Nucl. Soc.; 32: 537-539(Jun 1979). (CONF-790602--(Summ.)).

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

LMFBR TYPE REACTORS;TRANSIENTS;HYDRAULICS; HEAT TRANSFER;C CODES;COMPUTER CODES;REACTOR CORES

3226 TRANSIENT THERMAL-HYDRAULIC ANALYSIS IN THE FORCED-CONVECTION REGIME (TRANSENERGY). Glazer, S.; Greene, T.; Todreas, N.; Wolf, L. (Massachusetts Institute of Tech., Cambridge). Trans. Am. Nucl. Soc.; 32: 539-540(Jun 1979). (CONF-790602--(Summ.)).

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

LMFBR TYPE REACTORS;TRANSIENTS;HYDRAULICS; HEAT TRANSFER;FORCED CONVECTION;COMPUTER CODES; T CODES

3227 EFFECT OF FUEL DESIGN AND MATERIAL ON LMFBR TERMINATED TRANSIENT OVERPOWER PERFORMANCE. Coffield, R.D.; Markley, R.A.; Lake, J.A.; Dickson, P.W. (Westinghouse Advanced Reactors Division, Madison, PA). Trans. Am. Nucl. Soc.; 32: 576-578(Jun 1979). (CONF-790602--(Summ.)).

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

LMFBR TYPE REACTORS;TRANSIENT OVERPOWER ACCIDENTS;FUEL ELEMENTS;REACTOR CORES;DESIGN; THERMAL CONDUCTIVITY;TEMPERATURE DISTRIBUTION

3228 SCRAM SIGNAL GENERATOR. Johanson, E.W.; Simms, R. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 32: 623-625(Jun 1979). (CONF-790602--(Summ.)).

From ANS annual meeting; Atlanta, GA, USA (3 Jun 1979).

LMFBR TYPE REACTORS;FUEL MOTION DETECTION; FLOWMETERS;ELECTRONIC EQUIPMENT;SCRAM

3229 MODEL STUDIES OF BUBBLE SIZE DISTRIBUTION AND SOUND PROPAGATION AT MICROLEAKS IN SODIUM-HEATED STEAM GENERATORS. Uhlmann, G. (Zentralinstitut fuer Kernforschung, Rossendorf bei Dresden (German Democratic Republic)). Kernenergie; 22: No. 6, 198-205(Jun 1979). (In German).

The reaction zone of a small water leak in a sodium-heated steam generator (microleak) has been simulated by jet gassing or argon in water. The bubble diameter distribution in the bubble flow has been measured using a photoelectric method. The bubble size distribution obtained can be approached by an exponential distribution. For this case, phase velocity and sound damping have been calculated in the two-phase mixture. In the case of small ratios of sound frequency to the expected value of bubble resonance frequency, the frequency-

independent sound velocity of the homogeneous mixture is obtained as a function of the gas volume fraction. In the case of very high frequencies, the sound velocity of the pure liquid is obtained for any gas volume fractions. In the whole range investigated damping is strongly dependent on the frequency.

3230 (JAPFNR--458) FAST BREEDER REACTOR DEVELOPMENT IN JAPAN. PROGRESS REPORT, OCTOBER-DECEMBER 1978. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Jun 1979. Translation of PNC-N--251-79-04 17p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States-Japanese Fast Reactor Exchange Program.

Information is presented on the development of the experimental fast reactor JOYO; development of the fast breeder prototype reactor MONJU; design of a demonstration reactor; reactor physics; structural components; instrumentation and control; sodium technology; fuel materials; safety; and steam generator.

3231 THERMAL ANALYSIS OF AN IN-REACTOR LMFBR IRRADIATION CREEP EXPERIMENT. Feldman, E.E. (Argonne National Lab., IL (USA)). Nucl. Eng. Des.; 53: No. 1, 1-10(Jun 1979).

Thermal analyses of an in-reactor liquid metal fast breeder reactor (LMFBR) irradiation creep experiment and the test probe experiment which preceded it have been performed with the aid of a finite-difference heat transfer computer code. The calculated axial temperature distribution for the probe experiment is compared with the measured distribution. Although there is a 15% overprediction, the relative axial temperature distributions agree rather well. The thermal response of both experiments to a loss-of-flow transient is also analyzed.

3232 PROCESS AND APPARATUS FOR COOLING THE WALL OF THE VESSEL OF A FAST NEUTRON REACTOR. Andreani, A.; Chevallier, R.; Lambert, P. (to Commissariat a l'Energie Atomique). US Patent 4,156,629. 29 May 1979. Priority date 19 Apr 1977, France, 6p.

A process and apparatus for cooling the wall of the vessel of a fast neutron reactor comprising a main vessel which surrounds a second vessel containing the core and separating a volume of liquid metal contained in the main vessel into two regions called the hot collector and the cold collector respectively is described. The liquid metal drawn into the cold collector and passed through the core to the hot collector is circulated by means of pumps distributed around the core. A baffle and a counter-baffle are arranged concentrically to the main vessel wall and respectively define a first and second annular space issuing into the cold collector. The first annular space is traversed by a tapped flow of cooling liquid metal circulating from bottom to top in the first space in contact with the main vessel wall up to an upper constant level adjacent to the edge of the baffle. The liquid metal is poured into the second space up to a lower level corresponding to the free surface of the cold collector via helical channels in contact with the baffle wall directed towards the inside of the main vessel.

3233 CORE AND TRANSITION FUEL ASSEMBLY FOR A NUCLEAR REACTOR. Schreiber, R.E.; Sperhac, D.J. (to Westinghouse Electric Corp.). US

Patent 4,155,807. 22 May 1979. Filed date 22 Sep 1976. 10p.

A core and a transition fuel assembly for a nuclear reactor are described. They have a first fuel assembly including structure for laterally spacing parallel and coextending fuel rods positioned at preselected core elevations and also a second fuel assembly including lateral spacing structure at preselected core elevations at least one of which is different than the elevations of the spacing structure of the first fuel assembly. The transition fuel assembly is positioned between the first and second assemblies and includes lateral spacing structure positioned at each core elevation where the first and second fuel assemblies have a spacing structure. The transition fuel assembly ensures that contact among the fuel assemblies of the core is through the spacing structures. 9 claims.

3234 LIQUID METAL COOLED FAST BREEDER NUCLEAR REACTORS. Kenworthy, G.; Howard, R.S. (to Nuclear Power Co. Ltd.). US Patent 4,154,651. 15 May 1979. Filed date 2 Feb 1978. 4p.

In a liquid metal cooled fast breeder nuclear reactor of the pool kind the inner wall surface of the core tank has continuous fins inclined upwardly to form troughs through which cool liquid metal drawn from the outer region of the pool of coolant can cascade and thereby jacket the wall surface with relatively cool coolant.

3235 MAGNETIC NUCLEAR CORE RESTRAINT AND CONTROL. Cooper, M.H. (to Westinghouse Electric Corp.). US Patent 4,152,207. 1 May 1979. Filed date 20 Jan 1978. 6p.

A lateral restraint and control system for a nuclear reactor core adaptable to provide an inherent decrease of core reactivity in response to abnormally high reactor coolant fluid temperatures. An electromagnet is associated with structure for radially compressing the core during normal reactor conditions. A portion of the structures forming a magnetic circuit are composed of ferromagnetic material having a curie temperature corresponding to a selected coolant fluid temperature. Upon a selected signal, or inherently upon a preselected rise in coolant temperature, the magnetic force is decreased a given amount sufficient to relieve the compression force so as to allow core radial expansion. The expanded core configuration provides a decreased reactivity, tending to shut down the nuclear reaction.

3236 FRANCE SETS STANDARD FOR FAST REACTOR PROJECTS. Rippon, S. Energy Int.; 16: No. 5, 23-26(May 1979).

With their 1200-MW Super Phenix commercial fast reactor prototype under construction and the operational experience of a 250-MW experimental plant behind them, the French are acknowledged leaders in the development of these reactors. This has had a significant impact on other countries, and the adoption of the pool-type reactor in the United Kingdom and the Soviet Union makes it likely that this will be the basic type for all commercial fast-reactor plants in Europe, if not the world. U.S. policy, while delaying the commercialization of the fast breeder reactor, is maintaining a development program and could catch up if the political climate changes. A European license agreement is seen as the easiest way to launch a commercial fast reactor, however, and discussion between leading U.S. companies and the French have taken place.

3237 CHARACTERISTICS OF TRANSITION BOILING IN SODIUM-HEATED STEAM GENERATOR TUBES. France, D.M.; Carlson, R.D.; Chiang, T.; Priemer, R. (Argonne National Laboratory, Components Technology Division, Argonne, IL 60439). J. Heat Transfer; 101: No. 2, 270-275(May 1979).

Thermal fluctuations were measured in the tube wall in the transition boiling zone of a full-scale LMFBF sodium-heated steam generator tube. The tube had an inside diameter=10 mm, wall thickness=2.90 mm, heated length=13.1 m, and material=2 1/4 Cr-1 Mo steel. Water flowed vertically upwards inside the straight tube, and sodium flowed counter-currently in a surrounding annulus. Results of thermal, spectral, and thermal stress analyses are presented for a test within the normal operating range of LMFBF steam generators. Results of other tests are presented that show the effects and sensitivity of sodium temperature and water pressure on the severity of the thermal fluctuations.

3238 USE OF TEMPERATURE VARIANCE TO DETECT SODIUM BOILING. Meek, C.C. (Argonne National Lab., IL); Doerner, R.C.; Hurt, R.F. Nucl. Sci. Eng.; 70: No. 2, 201-204(May 1979).

Development of the temporal energy equations governing temperature variances has shown that abrupt increases in these quantities may be expected at coolant boiling inception. Such increases are dependent on local loss-of-coolant flow and the coupling between temperature and flow fluctuations. Observations from an in-pile liquid-metal fast breeder reactor simulation experiment demonstrate that variance increases can be appreciable.

3239 APPLICATION OF ACOUSTIC AGGLOMERATORS FOR EMERGENCY USE IN LIQUID-METAL FAST BREEDER REACTOR PLANTS. Shaw, D.T.; Rajendran, N. (State Univ. of New York, Buffalo). Nucl. Sci. Eng.; 70: No. 2, 127-134(May 1979).

The use of acoustic agglomerators for the suppression of sodium-fire aerosols in the case of a hypothetical core disruptive accident of a liquid-metal fast breeder reactor is discussed. The basic principle for the enhancement of agglomeration of airborne particles under the influence of an acoustic field is first discussed, followed by theoretical predictions of the optimum operating conditions for such application. It is found that with an acoustic intensity of 160 dB (approx. 1 W/cm²), acoustic agglomeration is expected to be several hundred times more effective than gravitational agglomeration. For particles with a radius larger than approx. 2 μ m, hydrodynamic interaction becomes more important than the inertial capture. For radii between 0.5 and 2 μ m, both mechanisms have to be included in the theoretical predictions of the acoustic agglomeration rate.

3240 FRENCH PROGRAM FOR ADVANCEMENT OF NUCLEAR POWER PLANTS WITH FAST BREEDERS. Vautrey, L. (CEA, 75 - Paris (France)). Kernenergie; 22: No. 5, 154-159(May 1979). (In German).

From 9. meeting on power plant engineering; Dresden, German Democratic Republic (25 - 26 Oct 1977).

A review is given of the development of fast reactors in France. The plants Rapsodie, Phenix and Superphenix, representing this development, are described, taking into consideration failures and difficulties hitherto observed.

3241 SENSITIVITY STUDY ON THE DAMAGE POTENTIAL OF A LIQUID-METAL FAST BREEDER REACTOR CORE DISASSEMBLY. Ostensen, R.W. (Sandia Labs., Albuquerque, NM). Nucl. Technol.; 43: No. 3, 301-313(May 1979).

In a core disruptive accident in a liquid-metal fast breeder reactor, rearrangement of core materials may lead to a prompt critical excursion. Damage to the primary containment, especially the head seals and head bolts, caused by the subsequent core disassembly is a primary concern of hypothetical accident analysis. A systematic study has been performed to evaluate the sensitivity of that damage in a commercial size reactor to various uncertainties. The damage is very insensitive to the equation-of-state of the fuel but highly sensitive to the reactivity ramp rate through prompt critical. From the point of view of vessel damage calculations, these results indicate that there is little incentive to improve our equation-of-state data on unirradiated mixed-oxide fuel.

3242 EFFECTS OF GRAIN BOUNDARY FISSION GAS ON TRANSIENT FUEL BEHAVIOR. DiMelfi, R.J.; Deitrich, L.W. (Argonne National Lab., IL). Nucl. Technol.; 43: No. 3, 328-337(May 1979).

The microstructural response of fast breeder reactor fuel to accident transients has been analyzed. Based on experimental results, fuel response can be classified as either basically brittle or basically ductile in nature. In the analysis, the type of response is assumed to be determined by the behavior of grain boundary fission gas. The transient variables taken into consideration are the temperature, heating rate, the mean gas content per bubble, mean bubble spacing in the grain boundary, and the stresses resolved normal to grain boundaries containing gas bubbles. By calculating the rate at which a grain boundary bubble grows as a sharp crack and comparing it to the rate of bubble growth by mass transport, a criterion is established to predict the characteristic response of a fuel sample to a specified thermal transient. A swelling threshold time is also determined for the case of ductile fuel behavior. Tensile stresses applied to the grain boundary are shown to enhance brittle behavior, and compressive stresses are shown to enhance ductile behavior. When average values of the relevant variables are extracted from a number of fission gas release and direct electric heating experiments and are used in the above calculation, fuel behavior predictions for these tests are found to correspond well with the experimental results.

3243 POSSIBILITY OF OBSERVING NEW MHD PHENOMENA IN THE LIQUID-METAL MASS IN THE FIRST LOOP OF THE BN-600 UNIT OF THE BELOYARSK ATOMIC POWER PLANT. Kirko, A.I.M.; Kirko, G.E. Sov. Phys. - Dokl. (Engl. Transl.); 24: No. 5, 399-400(May 1979).

Amplification of the magnetic field and the α -effect are discussed are discussed theoretically for the first loop of the fast-neutron reactor. (AIP)

3244 DEFORMATION ANALYSIS OF FUEL PINS WITHIN THE WIRE-WRAP ASSEMBLY OF AN LMFBR. Miki, K. (Hitachi Ltd., Ibaraki (Japan). Energy Research Lab.). Nucl. Eng. Des.; 52: No. 3, 371-384(May 1979).

Deformation of fuel pins within the wire-wrap fuel assembly of a fast breeder reactor is analyzed by two computational codes, the subchannel deformation analysis code SHADOW and the thermal-hydraulic analysis code DIANA. Coupling these codes makes it possible to

analyse precisely the mechanical interactions between all fuel pins in an assembly, and the deviation of coolant temperature distribution in deformed flow channels from the nominal distribution. In this paper, particular attention is paid to the effect on fuel pin deformation of the following factors; dimensional changes in the fuel assembly components, displacement of wrapper tube walls and changes in the radial power gradients.

3245 FAST-BREEDERS. THEIR CURRENT POSITION AND THEIR FUTURE PROSPECTS. Rozenholz, M. (Novatome Industries, 92 - Le Plessis-Robinson (France)); Megy, J. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Div. d'Etude et de Developpement des Reacteurs); Robert, E. (Societe Centrale Nucleaire Europeenne a Neutrons Rapides (NERSA), 69 - Lyon (France). Tech. Energ.; No. 26, 18-21(May 1979). (In French).

The introduction of fast-breeders in France should take place as follows: two plants of between 1200 and 1500 should be ordered at the beginning of 1980, 8000 to 10,000 megawatts of capacity should be installed by 1992, 25,000 megawatts by the year 2000. As it is very much a prototype, Creys-Malville will be relatively expensive (5.35 bln francs not counting the fuel). To get the cost down, one will have to count on the repetitive character of future construction programmes, on the simplification of the units and on technical progress. This anyway is the purpose of preparing the generation of reactors to follow Creys-Malville.

3246 NEW TYPE OF SIMULATOR, DEVELOPPED FOR SUPER-PHENIX BREEDER AND FOR PWR, ALLOWS PHENOMENA ANALYSIS AS WELL AS PROCESS CONTROL TRAINING. Bernard, P. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance; France). Dept. des Reacteurs a Eau); Quinton, J.-C. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance; France). Dept. des Reacteurs a Neutrons Rapides); Charton, H. (Compagnie Internationale de Services en Informatique, Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance; France)). Autom. Inf. Ind.; No. 77, 41-45(May 1979). (In French).

After an outline of the characteristics of study-type and training-type simulators, a new dual-purpose system is described which has been developed by the French CISI company, for a much lower cost than training simulator's one. For Super-Phenix reactor, the model describes the core, the primary vessel, reactor kinetics, the intermediate heat exchanger, the steam generator, piping and secondary tank, the turbine. For PWR, following components are modeled: core, steam generators, pressurizer, boiler. The control room includes two benchboards (trainee and instructor). A module design allows to move quickly from Super-Phenix configuration to PWR one.

3247 CONTRONIC 3 SYSTEM FROM HARTMANN AND BRAUN WILL ASSUME MEASUREMENT AND CONTROL FUNCTIONS OF THE SUPER-PHENIX REACTOR. Autom. Inf. Ind.; No. 77, 31(May 1979). (In French). AUTOMATION;ON-LINE CONTROL SYSTEMS;REACTOR CONTROL SYSTEMS;SUPER PHENIX REACTOR

3248 LIQUID METAL COOLED FAST BREEDER NUCLEAR REACTORS. Hopkins, S.R.; Diamond, A. (to Nuclear Power Co. Ltd., London (UK)). British Patent 2,005,901A/A/. 25 Apr 1979. 4p.

A liquid metal cooled fast breeder nuclear reactor comprises a core submerged in a pool of

liquid metal coolant in a primary vessel there being a fuel store disposed alongside the primary vessel and containing a second pool of liquid metal coolant. A rotary distributor disposed above and between the primary vessel and fuel store has complementary chutes which dip into the pools so that fuel carried in buckets can be transferred between the primary vessel and the fuel store. There is means for pumping coolant from each pool up to the distributor thence back to the pools by way of a distribution trough, tubes, a collecting trough and the chutes so that when irradiated fuel is being transferred from the fuel assembly to the store it is in contact with a continuously flowing stream of coolant.

- 3249 NUCLEAR POWER PLANT WITH EQUIPMENT FOR COOLANT CLEANING. Merkel, A. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,746,159/A/. 19 Apr 1979. 9p. (In German).

The cover gas argon which is located above the free surface of the sodium coolant, contains aerosols with Na, Cs 134 and Cs 137. In order to clean the cover gas and therefore the coolant, the cover gas with the aerosols is blown by a blower towards a vapour trap, which is provided with sintered metal bodies, where the liquid sodium is separated out. Subsequently the separated sodium is taken to a distillation device, where the Caesium is distilled off. The purified sodium then returns to the coolant circuit.

- 3250 AUXILIARY COOLING DEVICE FOR THE PRIMARY FLUID HEAT EXCHANGER OF A NUCLEAR REACTOR. Poudroux, P. (to Stein Industrie S.A.). US Patent 4,148,686. 10 Apr 1979. Priority date 18 Dec 1975, France, 4p.

An auxiliary cooling device for a primary fluid heat exchanger of a rapid neutron nuclear reactor is described, the exchanger having at least one nest of tubes in which there flows water to be heated to superheated steam and around whose tubes and inside whose casing there flows a liquid alkaline metal constituting the primary fluid. The casing of the exchanger is surrounded by a jacket having an air inlet tube and an air outlet tube at opposite ends. Inside the jacket there are auxiliary heat exchange surfaces such as spikes projecting from the casing of the primary heat exchanger proper. Application is to the absorption of residual power produced by a rapid neutron reactor after it is shut down.

- 3251 FAST REACTOR RECHARGING DEVICE. Artemiev, L.N.; Kurilkin, V.V. US Patent 4,147,588. 3 Apr 1979. Filed date 13 May 1977. 6p.

Disclosure is made of a device for recharging a fast-neutron reactor, intended for the transfer of fuel assemblies and rods of the control and safety system, having profiled heads to be gripped on the outside. The device comprises storage drums whose compartments for rods of the control and safety system are identical to compartments for fuel assemblies. In order to store and transport rods of the control and safety system from the storage drums to the recharging mechanism provision is made for sleeve-type holders. When placed in such a holder, the dimensions of a rod of the control and safety system are equal to those of a fuel assembly. To join a holder to a rod of the control and safety system, on the open end of each holder there is mounted a collet, whereas on the surface of each rod of the

control and safety system, close to its head, there is provided an encircling groove to interact with the collet. The grip of the recharging mechanism is provided with a stop interacting with the collet in order to open the latter and withdraw the safety and control system rod from its holder.

- 3252 DIFFUSION COEFFICIENTS OF SODIUM VAPORS IN ARGON AND HELIUM. Kumada, T.; Ishiguro, R.; Kimachi, Y. (Hokkaido Univ., Sapporo, Japan). Nucl. Sci. Eng.; 70: No. 1, 73-81 (Apr 1979).

Diffusion coefficients of sodium vapors in argon and helium gases were measured from 380 to 560°C by the Stefan method under atmospheric pressure. Careful considerations were made in the design of the apparatus to obtain data within a $\pm 10\%$ error, which arises from several sources inherent in the Stefan method. The experimental errors inherent in such sources were theoretically evaluated. The experimental diffusion coefficients were compared with those predicted by the majority of previous theoretical expressions. The comparisons revealed that the expressions proposed by Moulart for the sodium-argon mixture and by Aref'yev et al. for the sodium-helium mixture predicted values that were in excellent agreement with the measurements.

- 3253 WELDING THE CLINCH RIVER BREEDER REACTOR CORE SUPPORT STRUCTURE. Canary, W.W.; Franco-Ferreira, E.A. (Allis-Chalmers, York, PA). Weld. J. (Miami); 58: No. 4, 15-19 (Apr 1979).

The following accomplishments were realized in fabricating the CRBR core support structure: tools were obtained which successfully removed slag and ground starts and stops in a deep, narrow weld groove, dye penetrant examinations were successful and routinely performed in a deep, narrow weld groove, a single weld bead per layer technique was successfully used with no requirement for interpass grinding; a minimum amount of welding consumables and arc time was used in fabrication; a 24 in. (0.61 m) thick Type 304 stainless steel weld was qualified and a 21 in. (0.53 m) thick weldment was successfully made in production using the SAW process in a narrow groove; the CSP welding was completed without the requirement for any rework; the fabrication of the CSP was completed 2 1/2 months ahead of schedule; module liners were welded to close dimensional control and stringent inspection criteria, and a final assembly weld was made on final-machined components which met both stringent dimensional and inspection criteria.

- 3254 MECHANICAL BEHAVIOR OF HONEYCOMB SPOT-WELDED SPACER GRIDS IRRADIATED IN TWO FUEL ROD BUNDLES. Haas, D. (Belgonucleaire, Brussels, Belgium); Van de Velde, J.; Braun, H. Nucl. Technol.; 43: No. 1, 100-108 (Apr 1979).

In the Rapsodie-I experiment, two bundles, each containing 34 fast breeder reactor fuel pins, have been irradiated up to a peak burnup of 10.6 at. % and to a peak fast fluence ($E > 0.1$ MeV) of 6.65×10^{26} n/m². One of the main objectives of this experiment was to evaluate the mechanical behavior of a bundle with spacer grids. Two types of spacer grid designs have been tested: namely, a brazed ferrule grid design and a honeycomb spot-welded grid design. The grid material was in every case niobium-stabilized austenitic stainless steel type W.Nr.1.4981 in the annealed condition. The density and the dimensional measurements carried out on the spacer grids revealed that the geometrical changes in the grids were almost entirely due to material void swelling. In some cases, however, mechanical interactions

between grids and wrapper tubes and also between fuel pins and grid cells have been emphasized. These interactions had no detrimental influence on the in-pile bundle behavior. The postirradiation mechanical tests carried out on the honeycomb spacer grids showed that the mechanical properties of the grid cells have not been significantly altered by the irradiation. A decrease of the grid material Young's modulus has been correlated with void swelling. It has been concluded that the spacer grids operated satisfactorily despite their severe loading conditions.

3255 OVERVIEW OF FUEL ELEMENT DESIGN.

Olander, D.R. (Univ of Calif, Berkeley). Mech. Eng.; No. 4, 30-36(Apr 1979).

To sustain an adequate fission rate for optimum operation, the neutron flux in a fast reactor (LMFBR) must be 100 times larger than in a thermal or light water reactor. As a consequence, the LMFBR sustains far greater damage to its structural materials. The nature of this damage--the physics, chemistry, and mechanics of the processes occurring in the fuel and cladding--must be understood before commercialization of the breeder reactor can become a reality. This means that a successful fuel element design must be able to affirm with quantitative confidence that a potential design will remain intact for the reactor exposure to which it is to be subjected.

3256 NUCLEAR POWER TODAY AND TOMORROW.

Brown, K.T. (Atomic Energy Board, Pretoria, South Africa). Energy Environ.; 5: No. 4, 11, 15, 17-18(Apr 1979).

With this article the author wants to present a realistic assessment of the contribution which nuclear power is making now, and can make in the future, towards satisfying global energy demands. This paper also compares the possible effects of US policy on nuclear fuel resources (if they are generally adopted) with the former trend of the industry. He then comes to the following conclusions: (a) The future course of nuclear power will be largely determined by evolving national and international politics; (b) Recycling of plutonium in LWR's could relieve demands on uranium production, but could also impair the availability of plutonium for LMFBR's. (c) Current commercial systems will probably be adequate to fulfil predicted demand beyond the year 2000 in the Western world as a whole, but individual countries will experience uranium shortages then unless advanced systems can be developed. (d) The LMFBR is further advanced than any alternative type of breeder, and should be actively developed if it is to provide insurance as one possible means of helping to meet mankind's long term energy requirements. (e) There are many uncertainties involved in the development of an alternative fuel cycle, and the ability of any new system based on such a cycle to meet projected demands is also dubious.

3257 PROCEDURES USED IN THE UK FOR THE CERAMOGRAPHIC EXAMINATION OF (UPU)₂O₂ OXIDE FUELS FOR THE DOUNREAY PROTOTYPE FAST REACTOR.

James, G. (UKAEA, Windscale, Nuclear Power Development Labs.). J. Nucl. Mater.; 81: No. 1-2, 275-281(Apr 1979).

Ceramographic information on PFR fuel forms part of a comprehensive data bank established to record significant parameters on fuel, cladding and component materials loaded into the reactor. Collection of the data is essential for assessment of the effects of fuel and material variables on fuel element

performance. The methods developed for conversion of ceramographic data into computer compatible form for storage on the data bank, are described. The uniform code of practice used by the various laboratories involved in the manufacture of fuels for the PFR is explained and the methods by which the ceramographic information is fed into the data bank are indicated. The autoradiographic method used to establish plutonium distribution is discussed and the results compared with those from electron microprobe examination of unirradiated (UPu)₂O₂ specimens. Autoradiographic work on irradiated (UPu)₂O₂ is also discussed. Examples of Pu re-distribution are given and the results compared with those from EPMA examination of the specimens.

3258 (JAPFNR--460) PROGRESS REPORT ON FAST BREEDER REACTOR DEVELOPMENT IN JAPAN, JULY-SEPTEMBER 1978. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Apr 1979. Translation of PNC-N--251-79-02 16p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States - Japanese Fast Reactor Exchange Program.

Information is presented concerning development of the JOYO reactor; development of the MONJU reactor; reactor physics; structural components for the JOYO and MONJU reactors; instrumentation and control; sodium technology; fuel materials; structural materials; safety; and steam generator.

3259 STEEL Z6 CNTDV 25-15 FOR NUCLEAR BOLTS. DEVELOPMENT OF A TREATMENT ELIMINATING STRUCTURAL RETRACTION IN SERVICE AT HIGH TEMPERATURE. Cozar, R.; Julien, M. (Societe Creusot-Loire, 75 - Paris (France)); Gelpi, A. (Novatome Industries, 92 - Le Plessis-Robinson (France)). Mem. Sci. Rev. Metall.; 76: No. 4, 241-252(Apr 1979). (In French).

From 17. Meeting on alloy steels; Saint-Etienne, France (25 - 26 May 1978).

Two heat treatment were studied for the prevention of structural retraction produced under service conditions with conventional tempering and revealed by an increase in stress during the relaxation tests. The observed results are discussed.

3260 CORE CATCHER TUB WITH THERMAL INSULATION. Parr, A.; Kleffner, H.; Leder, H. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,741,795/A/. 29 Mar 1979. 22p. (In German).

The tub collects the sodium coolant in case of a Bethe-Tait accident. It consists of a steel shell having got an internal layer of blocks from Al₂O₃ with an addition of Na₂O and traces of Si. The blocks are superposed and held together in a support structure of double-T glands, independent of the steel shell and geared with each other in the way of building blocks. The gaps are filled with a fibrous material or sheets. Support structure and blocks having got different coefficients of expansion, the support structure may stand on a support ring divided into several sectors. The sectors are arranged on an intermediate ring by means of rolls and can independently from each other be moved in radial direction. The intermediate ring itself is mounted by means of rolls, permitting radial motion, on a base ring provided with stops and fastened to the support structure.

3261 METHOD AND CIRCUIT ARRANGEMENT FOR THE

DETECTION OF COOLING DISTURBANCES. Edelmann, M. (to Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,742,396/A/. 29 Mar 1979. 13p. (In German).

The circuit arrangement permits the detection of local cooling disturbances at the fuel element by measuring the fuel element outlet temperature of the coolant. For this purpose the operational variations of the measured signal lying in the low-frequency range are eliminated. The circuit consists basically of three measuring channels for neutron flux, inlet temperature of the coolant, and the coolant flow rate, the long-time average values of which are subtracted from the actual values of the coolant outlet temperature.

3262 NUCLEAR REACTOR. Jolly, R. (to UKAEA London Office; Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,839,211/A/. 15 Mar 1979. 20p. (In German).

The support grid for the fuel rods of a liquid metal cooled fast breeder reactor has a regular hexagonal contour and contains a large number of unit cells arranged honeycomb fashion. The totality of these cells make up a hexagonal shape. The grid contains a number of strips of material, and there is a window in each of three sidewalls staggered by one sidewall. The other sidewalls have embossed protrusions, thus generating a guide lining or guide bead. The windows reduce the rigidity of the areas in the middle between the ends of the cells.

3263 EMERGENCY SCRAM ACTUATION DEVICE FOR NUCLEAR REACTORS. Noyes, R.C.; Zaman, S.U.; Stuteville, D.W. (to Combustion Engineering, Inc., Windsor, CT (USA); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,831,028/A/. 15 Mar 1979. 12p. (In German).

The safety parameter employed for emergency scrams of a liquid metal cooled reactor is the coolant pressure. An actuation bellows is provided which is connected to a measuring chamber by means of a flow system. Both units are installed in a coolant flow section. The measuring chamber proper is connected with the coolant by means of an aperture limiting the flow. Inside the measuring chamber there is an expansion space filled with gas. Pressure changes in the coolant affect the pressure in the expansion space. Expansion of the bellows actuates the release mechanism.

3264 UV VIEWING THROUGH SODIUM COOLANT. Minnick, L.E.; Pack, R.W. (to Electric Power Research Inst.). US Patent 1980,001. 6 Mar 1979. Filed date 29 Jun 1977. 3p.

Sodium-immersed components in a nuclear reactor are rendered observable by illumination with radiation in the ultraviolet (UV) spectrum and by detecting reflected radiation through a UV scanning means, such as a television type camera sensitive to UV radiation. The UV scanning means may be adapted to be immersed in molten sodium coolant to facilitate observation of selected locations within the nuclear reactor. Real-time images may be provided for monitoring reactor operation and remotely-controlled reactor maintenance operations.

3265 DEVICE FOR PROVIDING A LEAK-TIGHT PENETRATION FOR ELECTRIC CABLES THROUGH A REACTOR VAULT ROOF. Eyral, M.; Mahe, A. (to

Commissariat a l'Energie Atomique). US Patent 4,142,937. 6 Mar 1979. Priority date 28 Jun 1976, France, 8p.

The device for providing a cable penetration through the vault roof of a liquid sodium cooled fast reactor comprises a vertical tube closed at the top end by a flange-plate. Electric cables connected to measuring and detecting instruments are passed through the flange-plate which is joined to the reactor vault roof in leak-tight manner and enclosed within a removable hood. At least one horizontal plate is mounted within the vertical tube and provided with orifices for the leak-tight passage of the cables. Cable storage reels are placed within the tube and can be locked in position or released by controlled mechanical means.

3266 SPACER FOR SUPPORTING FUEL ELEMENT BOXES. Wild, E. (to Gesellschaft fuer Kernforschung m.b.H.). US Patent 4,142,934. 6 Mar 1979. Priority date 14 Dec 1976, German, Federal Republic of (F.R. Germany), 4p.

A spacer plate unit arranged externally on each side and at a predetermined level of a polygonal fuel element box for mutually supporting, with respect to one another, a plurality of the fuel element boxes forming a fuel element bundle, is formed of a first and a second spacer plate part each having the same length and the same width and being constituted of unlike first and second materials, respectively. The first and second spacer plate parts of the several spacer plate units situated at the predetermined level are arranged in an alternating continuous series when viewed in the peripheral direction of the fuel element box, so that any two spacer plate units belonging to face-to-face oriented sides of two adjoining fuel element boxes in the fuel element bundle define interfaces of unlike materials.

3267 NUCLEAR REACTOR. Wade, E.E. (to Westinghouse Electric Corp.). US Patent 4,142,935. 6 Mar 1979. Filed date 9 Apr 1975. 8p.

Disclosed is a nuclear reactor including two rotatable plugs and a positive top core holddown structure. The top core holddown structure is divided into two parts: a small core cover, and a large core cover. The small core cover, and the upper internals associated therewith, are attached to the small rotating plug, and the large core cover, with its associated upper internals, is attached to the large rotating plug. By so splitting the core holddown structures, under-the-plug refueling is accomplished without the necessity of enlarging the reactor pressure vessel to provide a storage space for the core holddown structure during refueling. Additionally, the small and large rotating plugs, and their associated core covers, are arranged such that the separation of the two core covers to permit rotation is accomplished without the installation of complex lifting mechanisms.

3268 HYDROGEN CONCENTRATION METER UTILIZING A DIFFUSION TUBE COMPOSED OF 2 1/4 CR-1 MO STEEL AND A SLEEVE OF NICKEL. Roy, P.; Sandusky, D.W.; Hartle, R.T. (to Dept. of Energy). US Patent 4,143,316. 6 Mar 1979. Filed date 7 Sep 1977. 4p.

PAT-APPL-831,191.

A diffusion tube hydrogen meter for improving the sensitivity and response time for the measurement of hydrogen in liquid sodium is described. The improved hydrogen meter has a composite membrane composed of pure nickel

sleeve fitted, for example, over a 2 1/4 Cr-1 Mo steel or niobium diffusion tube. Since the hydrogen permeation rate through 2 1/4 Cr-1 Mo steels is a factor of four higher than pure nickel, and the permeation rate of hydrogen through niobium is two orders of magnitude greater than the 2 1/4 Cr-1 Mo steel, this results in a decrease in response time and an increase in the sensitivity.

- 3269 INTERCOMPARISON OF AUTO- AND CROSS-POWER SPECTRAL DENSITY SURVEILLANCE SYSTEMS FOR SODIUM BOILING DETECTION IN FAST REACTORS. Ehrhardt, J. (Kernforschungszentrum, Karlsruhe, Germany). Nucl. Technol.; 42: No. 3, 250-263(Mar 1979).

Theoretical and experimental investigations on detection systems for small narrow-band components in noise signals were conducted. These detection systems are based on the continuous surveillance of the power spectral density for characteristic peaks. Detection sensitivity for auto- and cross-correlation measurements was computed for signals with normally distributed amplitudes in dependence of signal coherence. The derived detection criteria allowed the comparison of auto- and cross-power spectral density surveillance. Theoretical results were confirmed in a number of experimental parameter studies. Special theoretical investigations were done for the optimal detection of local sodium boiling in liquid-metal fast breeder reactors.

- 3270 BUBBLE WORTH VARIATION IN MOLTEN CORES FROM A HYPOTHETICAL CORE DISRUPTIVE ACCIDENT. Marotta, C.R. (U.S. Nuclear Regulatory Commission, Washington, DC). Nucl. Technol.; 3: No. 3, 350-352(Mar 1979).

A simple and accurate calculation method is presented for the variation of bubble worth in a molten core. The method depends on previous analysis of reactivity increase due to compaction of unmoderated fissile systems. The average density of the fuel material after it has expanded into the bubble region is the controlling parameter of the proposed method. Comparison with detailed bubble worth calculations due to Nicholson and Goldsmith shows excellent agreement. Given a quality base worth case, the bubble worth variation for the perturbed system (of fuel, either moving into or out of the bubble volume) can be accurately calculated in terms of the base worth value. The proposed method can be used to evaluate the physical reasonableness of complex calculations using transport theory or Monte Carlo for estimates of bubble worth.

- 3271 CHARACTERIZATION OF FRAGMENTED FUEL FROM A THERMAL FUEL-SODIUM INTERACTION. Gross, S.; Vollath, D. (Kernforschungszentrum, Karlsruhe, Germany). Nucl. Technol.; 42: No. 3, 264-271(Mar 1979).

Out-of-pile experiments were performed on fuel configurations, having a geometry similar to that of reactor fuel elements to study the thermal interaction between molten UO_2 and subcooled sodium. The fragmented fuel generated was investigated by means of a scanning electron microscope. The composition of the fuel particles and the test results, especially the pressure pulses measured are analyzed.

- 3272 EFFECT OF AXIAL STRESS ON THE TRANSIENT MECHANICAL RESPONSE OF 20%, COLD-WORKED TYPE 316 STAINLESS-STEEL CLADDING. Yamada, H. (Argonne National Lab., IL). Nucl. Technol.; 42: No. 3, 324-331(Mar 1979).

To understand the effects of the fuel-

cladding mechanical interaction on the failure of 20% cold-worked Type 316 stainless-steel cladding during anticipated nuclear reactor transients, the transient mechanical response of the cladding was investigated using a transient tube burst method at a heating rate of 5.6°C/s and axial-to-hoop-stress ratios in the range of 1/2 to 2. The failure temperatures were observed to remain essentially constant for the transient tests at axial-to-hoop-stress ratios between 1/2 and 1, but to decrease with an increase in axial-to-hoop-stress ratios above unity. The uniform diametral strains to failure were observed to decrease monotonically with an increase in axial-to-hoop-stress ratio from 1/2 to 2, and in general, the uniform axial strains to failure were observed to increase with an increase in axial-to-hoop-stress ratio. The fracture of the cladding during thermal transients was found to be strongly affected by the maximum principal stress but not by the effective stress.

- 3273 TRITIUM PERMEATION THROUGH FE-2 1/4 CR-1 MO STEAM GENERATOR MATERIAL. Renner, T.A.; Raue, D.J. (Argonne National Lab., IL). Nucl. Technol.; 42: No. 3, 312-319(Mar 1979).

The permeability of tritium through Fe-2 1/4 Cr-1 Mo steel (Croloy) steam generator material has been measured for the surface-clean metal and for the metal oxidized in steam at 755 K (482°C). The temperature dependence of tritium permeability has been measured for the unoxidized metal between 673 and 773 K; the apparent activation energy for permeation in this temperature range was found to be 39.14 ± 0.32 kJ/mol (9350 ± 80 cal/mol). After steam oxidation at 755 K and 0.2 MPa (2 atm) for about one month, the tritium permeation rate decreased by a factor of approx. 150 relative to the clean metal. This reduction factor agrees very well with that measured for a sample of tubing used in Atomic International's Modular Steam Generator experiments under typical liquid-metal fast breeder reactor steam generator conditions. These results may be used to calculate rates of tritium transport from the intermediate heat transport system sodium through the steam generator tubes and into the steam/water system. From this information, eventual release rates to the environment can be determined.

- 3274 FILTRATION OF SODIUM-FIRE AEROSOLS.

Alexas, A.; Jordan, S.; Lindner, W. (Kernforschungszentrum, Karlsruhe, Germany). Nucl. Technol.; 42: No. 3, 320-323(Mar 1979).

Different filter devices have been developed and tested with respect to their use in the off-gas system of liquid-metal fast breeder reactors to prevent the escape of sodium-fire aerosols that might be formed in case of an accident. The testing results have shown that the use of a multilayer sand bed filter is still the best method to filter limited amounts of sodium-fire aerosols over a long operating time. Efficiencies on the order of 99.98 and 98.8% were reached for loading capacities of 500 and 1000 g/m², respectively. Unlimited amounts of sodium-fire aerosols can be filtered by wet scrubbers with an efficiency of 70% per scrubber stage. Fiberglass filters cannot be used for the filtration of sodium-fire aerosols over a long operating time because the filter material can be destroyed after several days of operating.

- 3275 CORROSION BEHAVIOR OF SODIUM-EXPOSED STAINLESS STEELS IN CHLORIDE-CONTAINING AQUEOUS SOLUTIONS. Wu, P.C.S.; Grundy, B.R.; Miller, R.L. (Westinghouse Advanced Reactors

Division, Madison, PA). Metall. Trans., A; 10: No. 3, 319-325 (Mar 1979).

The corrosion behavior of sodium-exposed stainless steels in chloride-containing aqueous solutions was investigated. Results showed that sodium-corroded Type 316 stainless steel (prototypic Liquid Metal Fast Breeder Reactor (LMFBR) fuel cladding) maintains its integrity after five months exposure in these solutions at 82°C and with chloride content up to 500 ppm. In contrast, sensitized and sodium mass transfer deposit-containing Type 304 stainless steel failed in the high chloride solution (500 ppm) within ten days at the same temperature. The failure was initiated by pitting and subsequently accelerated by intergranular attack. The results also show that high pH tends to reduce the susceptibility to failure while procedures commonly used for sodium removal have no significant effect on the water corrosion behavior of the test material. Based on the current results, it is concluded that water shortage is feasible for spent fuels in a LMFBR reprocessing plant.

3276 HETEROGENEOUS CORES FOR LMFBR IMPROVES BREEDING. Nucl. Eng. Int.; 24: No. 282, 63 (Mar 1979).

BREEDING; BREEDING BLANKETS; FUEL ASSEMBLIES; LMFBR TYPE REACTORS; PLUTONIUM; REACTIVITY; REACTOR CORES; DESIGN; SODIUM; VOIDS

3277 PHYSICS PROBLEMS OF FAST REACTOR SHIELDING. Farinelli, U. (Comitato Nazionale per l'Energia Nucleare, Casaccia (Italy). Centro di Studi Nucleari); Nicks, R. (Commission of the European Communities, Ispra (Italy). Joint Research Centre). At. Energy Rev.; 17: No. 1, 129-179 (Mar 1979).

The physics problems associated with the design of neutron and gamma shields for Liquid Metal Fast Breeder Reactors are reviewed. The main objectives of shielding in terms of biological protection, reduction of activation, radiation damage and nuclear heating are discussed in terms of the accuracies required for the design. Methods used to calculate the sources and the transport of neutrons and gammas are discussed in the light of the most recent developments and in view of their application to specific fast reactor problems; streaming problems tend to be more exacting than those associated with bulk shielding. Within the frame of the dominating transport $S_{sub}(n)$ and Monte Carlo approaches several new options are becoming available, while alternative methods continue to play an important role. Great progress has recently been achieved in obtaining nuclear data for fast reactor shielding applications; although a still better accuracy is required, the prospectives of obtaining an adequate and consistent data base are encouraging. Sensitivity studies and the assessment of uncertainties and covariances in nuclear data play an important part in this process. Integral experiments are used to improve, validate or assist calculations but very seldom to replace them. Simple and accurate experiments of a benchmark nature (such as neutron propagation in blocks of single materials) are currently being used to assess or improve the accuracy of nuclear data or calculational methods. Finally, calculations and experiments related to some actual and typical design problems are reported.

3278 INFLUENCE OF PHYSICO-CHEMICAL FORM ON THE CONSEQUENCES OF HYPOTHETICAL ACCIDENTAL RELEASES FROM FBRS. Kelly, G.N. (National Radiological Protection Board, Harwell (UK)).

Radiol. Prot. Bull.; No. 27, 12-14 (Mar 1979).

A summary is given of a recent Report (Kelly, G.N., Jones, J.A. and Simmonds, J., The influence of the physico-chemical form of the aerosol on the radiological consequences of notional accidental releases of radioactivity from a fast breeder reactor, NRPB-R73. London, HMSO (1979)). The results are of interest in view of the possibility of a release of a mixed sodium-transuranium element aerosol from a LMFBR; in this form the transuranium elements once inhaled, may be more readily transported to other body organs, thus increasing the doses. Consideration has been given to the sensitivity of the consequences of releases of activity to a wide range of physico-chemical forms of the aerosol (0.1 μ m activity median aerodynamic diameter (AMAD) to 10 μ m AMAD; mixed sodium-element aerosol or homogeneous aerosol of the same chemical composition as the fuel). The variation in biological effect with physico-chemical form was shown to be sufficiently low to justify the use of the 'reference' (1 μ m AMAD, oxide form) aerosol for generic assessments of releases from FBR, except in some estimates of the incidence of early morbidity and to a lesser extent early mortality resulting from large releases.

3279 LIQUID METAL COOLED FAST REACTORS DEVELOPMENT IN FRANCE. Bogatyreva, T.N. (Energomashinostroenie; No. 3, 39-42 (Mar 1979). (In Russian).

A brief history of LMFBR reactor development in France has been stated. Described are specifications and the experience of operation of 24 MW Rapsodie sodium-cooled reactor (1960 - 1967) and 250 MW Phenix sodium-cooled reactor (1965 - 1973) as well as characteristics of 1200 MW Phenix reactor under construction. It is shown that France takes the first place in the world in this field of atomic power industry due to great progress in development and construction of fast breeder reactors with liquid-metal coolant. The Super-Phenix project is now supposed to improve and on its base to create facilities of 1800-1900 MWe. Creation of 40 MWe experimental carbide fuel fast reactor is being planned in 1980-1984 within the framework of the plan of perspective improving fuel cycle.

3280 TEMPERATURE FLUCTUATIONS: AN ASSESSMENT OF THEIR USE IN THE DETECTION OF FAST REACTOR COOLANT BLOCKAGES. Greef, C.P. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Nucl. Eng. Des.; 52: No. 1, 35-55 (Mar 1979).

The temperature noise technique for the detection of local blockages in fast reactor subassemblies is discussed. The main factors involved in an assessment of the technique are outlined and the experimental and theoretical work that has been carried out at BNL on the various aspects of the problem is described. It is concluded that blockings appreciably smaller than those predicted to produce boiling should be detectable against a background noise level due to subassembly power tilts, on a time scale giving protection against rapidly developing incidents. Further work required to increase confidence in the application of the technique to the reactor is outlined, including measurements in fully representative geometries, data from sodium rigs and further information on reactor background noise levels.

3281 HEAT TRANSFER EFFECT OF ENTRAINED GAS IN LIQUID SODIUM SYSTEMS. Bishop, A.A. (Pittsburgh Univ., PA (USA)); Engel, F.C.; Markley, R.A. (Westinghouse Electric Corp.,

Pittsburgh, PA (USA). Advanced Reactors Div.). Nucl. Eng. Des.; 52: No. 1, 1-13(Mar 1979).

An analysis was made and a correlation developed to determine the reduction in heat transfer caused by entrained inert gas in sodium-cooled liquid metal systems. A 1% void fraction (1.62% volumetric flow rate of inert gas) is calculated to reduce the heat transfer coefficient by approximately 4% in a typical LMFBR. This reduction will not occur over the entire reactor, but only near the inlet of the radial blanket. It was concluded that a reduction in heat transfer in a system having a low oxygen concentration (<5 ppm) occurs only if the heat transfer surface is unwetted or only partially wetted and entrained gas is present. However, if the surface temperature is greater than some 530°C (1000°F), if the Reynolds number is greater than approximately 10^5 , and if the operating time is longer than that required for complete wetting at the reference temperature, no reduction in heat transfer is expected. The cause of the reduction in heat transfer is postulated to be a reduction in the thermal conductivity of a two-phase sodium-gas layer near the heated surface. It is proposed to calculate the thermal conductivity reduction using the average inert gas void fraction and a sintered-metal model. (Auth.)

- 3282 SOME POST OPERATIONAL ADJUSTMENTS TO THE PROTOTYPE FAST REACTOR AT DOUNREAY. Lunt, A.R.W. (Nuclear Power Co. (Risley) Ltd. (UK)). J. Inst. Nucl. Eng.; 20: No. 2, 57-63(Mar-Apr 1979).

Prior to and during the initial operation of the Prototype Fast Reactor at Dounreay certain features have been considered to be in need of adjustment to provide better operating characteristics. This article describes the work done to support the consequential changes of operational techniques and plant design in the following areas: maintenance of dry conditions at the superheater steam inlets, the temperature control of the reactor roof, and the introduction of a system enabling the reactor to continue running after a turbine trip.

- 3283 CHEMICAL REACTIONS OF SODIUM FIRE AEROSOLS IN THE FREE ATMOSPHERE. Hofmann, Ch.; Jordan, S.; Lindner, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). J. Aerosol Sci.; 10: No. 2, 191-192(Mar 1979).

From Annual conference of the Gesellschaft fuer Aerosolforschung (GAF); Vienna, Austria (26 - 28 Sep 1978).

LMFBR; AEROSOLS; AIR; CHEMICAL REACTIONS; COMBUSTION; EXPERIMENTAL DATA; FIRES; GRAPHS; HUMIDITY; SODIUM; LMFBR TYPE REACTORS; REACTOR SAFETY; SODIUM OXIDES

- 3284 FIRST APPLICATION OF THE BAILEY CONTROL MICRO-Z SYSTEM. CONTROL OF SODIUM PURITY IN THE SUPER PHENIX REACTOR. Hureau, J.C. Autom. Inf. Ind.; No. 75, 35-39(Mar 1979). (In French).

The system for testing the purity of sodium in the Super Phenix reactor at Crey-Malville is one of the very first applications of the 'μz' Bailey Control system which, as is generally known, uses a microprocessor on each regulator. This characteristic is cleverly exploited here for discontinuous regulation, in stages, of an analog parameter: temperature, with a quite remarkable flexibility of use and an extremely reduced volume of equipment.

NEUTRON ABSORBING MASSES. Giuggio, N.; Noyes, R.C. (to Combustion Engineering, Inc.). US Patent 4,139,414. 13 Feb 1979. Filed date 19 Jun 1977. 6p.

An apparatus is disclosed for holding, releasing, and resetting a multiplicity of neutron absorbing balls within a safety vertically hinged trap doors resting on the shoulders of a generally cylindrical release valve, the actuation of which disengages the doors, permitting the poison balls above the doors to drop into the core. In the reset mode of operation a platform is raised, lifting the balls from the bottom of the core and swinging the trap doors upward until the balls are above the door hinges. The release valve is reset and the platform is lowered to reset the doors against the valve shoulders. In the disclosed embodiment, the valve is operated by a self-actuated, temperature responsive scram mechanism.

- 3286 MEASURING PROBE AND METHOD TO DETERMINE THE OXYGEN CONTENT OF GASES, VAPOURS AND LIQUIDS, IN PARTICULAR FOR LIQUID METALS. Sundermann, H.; Andrae, U. (to Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)); Deutsches Patentamt, Muenchen (Germany, F.R.)). German(FRG) Patent 2,061,599/C/. 8 Feb 1979. 4p. (In German).

The core of the measuring probe consists of a pot-shaped solid state electrolyte with a metal holder and a reference chamber with a constant oxygen partial pressure out of which the reference electrode is led. The reference electrode is surrounded by a cylindrical metal mesh at a present potential. The solid state electrolyte consists of yttrium oxide stabilized zirconium dioxide or thorium dioxide, the holder is made of a Fe-Co-Ni alloy, and the material closing the reference chamber is soft glass. The measuring probe lends itself particularly well to applications in liquid metal cooling systems.

- 3287 FUEL ROD FOR A SODIUM-COOLED REACTOR. Ernst, H. (to Kraftwerk Union A.G., Muelheim an der Ruhr (Germany, F.R.)); Deutsches Patentamt, Muenchen (Germany, F.R.)). German(FRG) Patent 2,032,030/B/. 8 Feb 1979. 4p. (In German).

The fuel element is filled with pellets, and the space between the cladding and the fuel contains sodium for sodium bonding purposes. Between the fuel and the upper opening there is a throttle followed by a sealing cone. The throttle is easily permeable only to gases. The carrier of the rod is equipped with a non-return valve which, under normal conditions, only allows sodium to pass from the outside into the rod. It contains an additional relief valve which allows sodium to escape from the rod, if the internal pressure is too high. Thus, in case of leaks the sodium bonding can be restored from the rod carrier.

- 3288 LAMINAR, TRANSITION, AND TURBULENT PARALLEL FLOW PRESSURE DROP ACROSS WIRE-WRAP-SPACED ROD BUNDLES. Engel, F.C. (Westinghouse Advanced Reactors Div., Madison, PA); Markley, R.A.; Bishop, A.A. Nucl. Sci. Eng.; 69: No. 2, 290-296(Feb 1979).

Laminar, transition, and turbulent parallel flow pressure drop across wire-wrapped hexagonal rod bundles positioned inside a duct were determined in tests using water, sodium, and air. A smooth transition region from turbulent to laminar flow that occurred over the Reynolds number range from 5000 to 400 characterized the resulting friction factor

- 3285 SCRAM DEVICE HAVING A MULTIPLICITY OF

behavior. The continuous transition region could be explained in terms of the fraction of the flow area in turbulent flow. Laminar friction factors calculated from individual subchannel measurements could be correlated by the same expression found for rod-bundle-averaged conditions. Friction factor correlation functions are given for flows in the laminar range, the turbulent range, and the transition range. A general laminar flow friction factor correlation was developed which agrees satisfactorily with limited laminar flow data from rod bundles having different wire-wrap lead pitch-to-diameter ratios.

- 3289 FAST INTEGRAL MODELING OF HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS IN LIQUID-METAL FAST BREEDER REACTORS. Mueller, C.J.; Vaurio, J.K. (Argonne National Lab., IL). Nucl. Sci. Eng.; 69: No. 2, 264-278 (Feb 1979).

The paper describes the basic equations and solution techniques of a collection of heat transfer and coolant voiding dynamics models that have been developed and successfully applied to simulate hypothetical accidents in liquid-metal-cooled fast breeder reactors (LMFBRs) to the point of permanent subcriticality or to the initiation of a prompt-critical excursion. These models emphasize analytic and integral solution techniques to minimize computational time and have been programmed into the SACO fast-running accident analysis computer code. The comparisons of SACO results to analogous SAS3D results used to qualify these models are illustrated and discussed. The fast-running nature of these models makes them an ideal sensitivity analysis tool for use in probabilistic evaluations of LMFBR accidents. Their use in this application is illustrated.

- 3290 MULTICHANNEL MODEL FOR RELOCATION OF MOLTEN FUEL CLADDING IN UNPROTECTED LOSS-OF-FLOW ACCIDENTS IN LIQUID-METAL FAST BREEDER REACTORS. Ishii, M.; Chen, W.L.; Grohmes, M.A. (Argonne National Lab., IL). Nucl. Sci. Eng.; 69: No. 2, 297-318 (Feb 1979).

During an unprotected undercooling accident in a liquid-metal fast breeder reactor, the motion and relocation of the molten fuel cladding can be important because of their potentially significant effect on reactivity, blockage formation, and subsequent fuel motion. The cladding relocation problem based on a multichannel film flow model is analyzed in the study. The important aspects considered in the analysis are the nonuniform transverse cladding melting pattern and sodium vapor flow diversions within a subassembly. It has been shown that the motion of molten fuel cladding and subsequent blockage formations can be significantly influenced by this interconnected channel effect. Several sample calculations have been made to demonstrate these points.

- 3291 REACTOR PHYSICS PARAMETERS OF ALTERNATE FUELED FAST BREEDER REACTOR CORE DESIGNS. Haffner, D.R.; Hardie, R.W. (Hanford Engineering Development Lab., Richland, WA). Nucl. Technol.; 42: No. 2, 123-132 (Feb 1979).

Nuclear nonproliferation considerations have resulted in renewed interest in the thorium fuel cycle. Reactor physics parameters of a typical 1200-MW(electric) fast breeder reactor design were compared for the cases when ^{233}U is substituted for plutonium as a fissile fuel and when ^{232}Th is substituted for ^{238}U as a fertile fuel. When the ^{238}U in the blanket is replaced with ^{232}Th , the reactor physics parameters are relatively unchanged. However, replacing ^{238}U in the core with ^{232}Th increases the critical

mass by 11 to 15% and decreases the breeding ratio by 0.13 to 0.16. In addition, replacing the plutonium in the core with ^{233}U decreases the critical mass by 4 to 6% and decreases the breeding ratio by 0.13 to 0.16. Both of the changes in the core make the sodium void coefficient more negative.

- 3292 PERFORMANCE OF THORIUM-FUELED FAST BREEDERS. Marr, D.R.; Cantley, D.A.; Chandler, J.C.; McCurry, D.C.; Omberg, R.P. (Hanford Engineering Development Lab., Richland, WA). Nucl. Technol.; 42: No. 2, 133-143 (Feb 1979).

Three studies were performed to evaluate the breeding ratio of fast breeders containing thorium. In a study of a small breeder, thorium metal and thorium oxide core designs were found to have similar breeding ratios. The slight advantage exhibited by the metal design was not considered significant, since the design was based on a limited amount of thorium metal swelling data. In a study of a 1200-MW(electric) plant, a plutonium-uranium oxide design was compared to a uranium-thorium metal design. The uranium-thorium design had a lower breeding ratio but also had a negative sodium void effect. In the third study, the effect of replacing thorium oxide radial blankets with thorium metal radial blankets was evaluated. This was found to have little effect on the breeding ratio.

- 3293 EFFECTS OF STOICHIOMETRY ON CLADDING ATTACK IN UO_2 - PuO_2 . Lawrence, L.A.; Hata, D.C.; Weber, J.W. (Hanford Engineering Development Lab., Richland, WA). Nucl. Technol.; 42: No. 2, 195-206 (Feb 1979).

A significant reduction in and a change in the character of fuel-cladding chemical interaction (FCCI) due to a reduction in the oxygen-to-metal ratio (O/M) were established for uranium-plutonium mixed-oxide fuels clad with 20% cold-worked Type 316 stainless steel irradiated in the Experimental Breeder Reactor II. Fuel pins from the Hanford Engineering Development Laboratory P-23C subassembly at initial fuel O/Ms of 1.94, 1.95, and 1.97 were examined at peak burnups of 1.1, 2.5, and 3.6 at.%. The depth of FCCI increased with increasing burnup and temperature, but the relative effects of fuel O/M did not change. An approximate three-fold reduction in FCCI resulted from a reduction in fuel O/M from 1.97 to 1.95 at approx. 3.6 at.% burnup. The peak FCCI in the lower O/M fuel was approx. 5 μm and appeared as a preferential loss of cladding at grain boundaries and slip planes on the cladding inner surface. In contrast, the typical FCCI in the higher O/M (1.97) fuel pin was matrix in character, and penetrated approx. 14 μm into the cladding. Thus, FCCI in mixed-oxide fuel can be reduced to negligible levels by lowering the O/M during manufacture of the fuel.

- 3294 SOME ELECTRON MICROPROBE ANALYSIS RESULTS CONCERNING THE INTERACTION BETWEEN URANIUM-PLUTONIUM MIXED-OXIDE FUEL AND STAINLESS-STEEL CLADDING. Walker, C.T.; Pickering, S. (European Inst. for Transuranium Elements, Karlsruhe, Germany). Nucl. Technol.; 42: No. 2, 207-215 (Feb 1979).

Analyses were performed on three mixed-oxide fuel pins. Two were irradiated in a fast flux, one in an epithermal-neutron flux. The compositions of the corrosion product phases in the fuel-cladding gaps of the different pins were similar. The phase was essentially a mixture of metal oxides, with chromium oxide the main constituent. Cesium chromate, if it

formed at all, was present in only small amounts. Oxides of iron and nickel were not detected, which suggests that the oxygen potential in the gap did not exceed that for the FeCr_2O_4 formation. Metallic fragments in the phase resulted from mechanical interactions involving the phase and cladding grains whose boundaries had been weakened by intergranular corrosion. Chromium and manganese were lost from the inner cladding surface of all three pins. Titanium loss also occurred from the two pins clad with titanium-stabilized steel. A grain boundary phase depleted in chromium was present at the inner cladding surface of one of the pins irradiated in a fast flux. The phase that was associated with intergranular attack occurred in advance of the corrosion front.

- 3295 EXPERIMENTAL PARAMETER INVESTIGATIONS ON THE TRIBOLOGICAL BEHAVIOR OF STELLITE 6 IN LIQUID SODIUM. Wild, E.; Mack, K.J. (Kernforschungszentrum Karlsruhe, Germany). Nucl. Technol.; 42: No. 2, 216-223 (Feb 1979).

The friction and wear behavior of various material combinations was studied in a liquid-sodium loop up to fluid temperatures of 900 K to guarantee self-alignment and safe operation of liquid-metal-cooled fast breeder reactor (LMFBR) components, such as subassemblies, control and shutdown units, instruments, and centering pads. Pin-on-disk, plate-on-plate, and tilting plane test section systems were used. Numerous material combinations exhibiting good corrosion resistances in liquid sodium were investigated under identical operating conditions. Those revealing the lowest wear rates and the lowest friction coefficients were subject to parameter tests. The influences on friction and wear behavior of friction velocity, contact force, sodium temperature, equipment criteria, and tribochemistry were studied. The experimental results are shown for a Stellite 6-Stellite 6 combination. They revealed friction and wear coefficients to be clearly below the set limits. However, this material combination has a high cobalt content and is therefore no optimum choice with respect to contamination problems of the primary system of an LMFBR. Other experiments are necessary to develop materials with zero contamination effects.

- 3296 FAST BREEDER. DO WE NEED IT. HOW DOES IT WORK. Hueper, R. Bild Wiss.; 16: No. 2, 71-85 (Feb 1979). (In German).

In Kalkar at the Lower Rhine, the German-Belgian-Dutch prototype of a fast breeder line is being built. This reactor type will generate power and turn non-fissile U^{238} into fissile Pu^{239} . The fast breeder thus makes much better use of natural uranium than normal nuclear power plants can. The breeder prototype at Kalkar was quite an issue in the nuclear power discussion: The liquid sodium used as heat transfer medium and the increased production of highly toxic plutonium require special safety measures in fast breeders.

- 3297 NUCLEAR INSTRUMENTATION CABLE END SEAL. Cannon, C.P.; Brown, D.P. (to Dept. of Energy). US Patent 4,137,423. 30 Jan 1979. Filed date 16 Dec 1977. 6p. PAT-APPL-861,132.

An improved coaxial end seal for hermetically sealed nuclear instrumentation cable exhibiting an improved breakdown pulse noise characteristic under high voltage, high temperature conditions is described. A tubular insulator body has metallized interior and exterior surface portions which are braze sealed to a center conductor and an outer

conductive sheath. The end surface of the insulator body which is directed toward the coaxial cable to which it is sealed has a recessed surface portion within which the braze seal material terminates.

- 3298 PRESSURE SENSOR. Forster, G.A. (to Department of Energy, Washington, DC (USA); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,824,794/A/. 25 Jan 1979. 11p. (In German).

The pressure sensor will be used for pressure measurement in high-temperature liquids, in particular for Na or K during fast breeder operation. It has a soft membrane which is unshielded at the pressure medium side and can actuate a rigid membrane on the other side by means of a bottom element. The displacement of the rigid membrane is proportional to the difference between the liquid pressure and a reference gas pressure. The displacement can be measured by means of a capacity displacement measuring instrument, using a fixed plate as reference. Absolute calibration is performed by admitting pressurized gas into the zone between the membranes and into the zone between rigid membrane and fixed plate. The coupling between the soft and the rigid membrane is thus interrupted.

- 3299 BRACING SYSTEM FOR THE FUEL ELEMENTS OF A REACTOR CORE. Class, G. (to Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,162,171/B/. 25 Jan 1979. 8p. (In German).

The bracing system for the prismatic fuel elements of a sodium cooled reactor core, for example, helps the structural dimensions. Therefore the fuel elements are connected together in each bracing plane by bearing elements such as balls, rolls, ball - or needle bearings or by elastic, deformable spring elements. For bracing on the circumference of the reactor core, there are also bracing elements of hexagonal cross-section which are divided longitudinally in the bracing area. By interaction with the elastic elements the fuel elements are held in the right geometric position in the hexagonal lattice in the bracing planes while they are pushed together.

- 3300 NUCLEAR CORE POSITIONING SYSTEM.

Garkisch, H.D.; Yant, H.W.; Patterson, J.F. (to Westinghouse Electric Corp.). US Patent 4,135,974. 23 Jan 1979. Filed date 17 May 1974. 8p.

Disclosed is a structural support system for the core of a nuclear reactor which achieves relatively restricted clearances at operating conditions and yet allows sufficient clearance between fuel assemblies at refueling temperatures. Axially displaced spacer pads having variable between pad spacing and a temperature compensated radial restraint system are utilized to maintain clearances between the fuel elements. The core support plates are constructed of metals specially chosen such that differential thermal expansion produces positive restraint at operating temperatures.

- 3301 SYSTEM FOR DETECTING THE FAILURE OF A NUCLEAR FUEL ROD IN A NUCLEAR REACTOR. Mitsutsuka, N.; Kubo, K.; Miyazawa, T. (to Tokyo Shibaura Electric Co., Ltd.). US Patent 4,135,970. 23 Jan 1979. Priority date 27 Feb 1976, Japan, 8p.

A system for detecting the failure of a nuclear fuel rod in a nuclear reactor comprises

sampling a coolant in a nuclear reactor, cooling the sampled coolant to permit ^{135}I in the coolant to be adsorbed on an iodine adsorption device, moving $^{135}\text{m}/\text{sub Xe/}$ produced by the ^{135}I decay into a cover gas at the upper space of a gas-liquid separator, and introducing the $^{135}\text{m}/\text{sub Xe/}$ entrained cover gas into a radioactivity detector where the radioactivity of $^{135}\text{m}/\text{sub Xe/}$ is counted.

3302 CORE DISRUPTIVE ACCIDENT MARGIN SEAL.

Golden, M.P. (to Dept. of Energy). US Patent 4,135,973. 23 Jan 1979. Filed date 10 Mar 1977. 8p.
PAT-APPL-776,328.

An apparatus for sealing the annulus defined within a substantially cylindrical rotatable riser assembly and plug combination of a nuclear reactor closure head is described. The apparatus comprises an inflatable sealing mechanism disposed in one portion of the riser assembly near the annulus such that upon inflation the sealing mechanism is radially actuated against the other portion of the riser assembly thereby sealing the annulus. The apparatus further comprises a connecting mechanism which places one end of the sealing mechanism in fluid communication with the reactor cover gas so that overpressurization of the reactor cover gas will increase the radial actuation of the sealing mechanism thus enhancing sealing of the annulus.

3303 SAFETY ARRANGEMENT FOR NUCLEAR REACTORS.

Giuggio, N.; Noyes, R.C.; Zaman, S.U. (to Combustion Engineering, Inc., Windsor, CT (USA); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,825,439/A/. 18 Jan 1979. 32p. (In German).

The temperature-dependent, automatic scram mechanism for liquid-metal cooled reactors uses the average outlet temperature of the cooling medium as critical quantity. It is determined by the interaction between the representative values of the flow- and throughput-amounts through a flow path of the cooling medium in a safety arrangement and the produced heat of the measurement fuel rod in this flow path. The outlet temperature is measured over a closed flow medium with a high expansion coefficient; its expansion is transformed in a linear force for the operation of the release mechanism. Temperature sensor, operation mechanism and safety rod are located in the safety arrangement.

3304 TECHNIQUE FOR DETECTING LIQUID METAL LEAKS.

Bauerle, J.E. (to Dept. of Energy). US Patent 4,134,290. 16 Jan 1979. Filed date 6 Sep 1977. 4p.
PAT-APPL-830,948.

In a system employing flowing liquid metal as a heat transfer medium in contact with tubular members containing a working fluid, i.e., steam, liquid metal leaks through the wall of the tubular member are detected by dislodging the liquid metal compounds forming in the tubular member at the leak locations and subsequently transporting the dislodged compound in the form of an aerosol to a detector responsive to the liquid metal compound. In the application to a sodium cooled tubular member, the detector would consist of a sodium responsive device, such as a sodium ion detector.

3305 PIPE SUPPORT.

Pollono, L.P. (to Dept. of Energy). US Patent 4,134,563. 16 Jan 1979. Filed date 28 Jan 1976. 4p.
PAT-APPL-653,044.

A pipe support for high temperature, thin-walled piping runs such as those used in nuclear systems is described. A section of the pipe to be supported is encircled by a tubular inner member comprised of two walls with an annular space therebetween. Compacted load-bearing thermal insulation is encapsulated within the annular space, and the inner member is clamped to the pipe by a constant clamping force split-ring clamp. The clamp may be connected to pipe hangers which provide desired support for the pipe.

3306 RADIATION SHIELDS.

Kanao, Y. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 54-3688/A/. 11 Jan 1979. 3p. (In Japanese).
A penetrating portion having a step-like cross section is provided for a radiation shield, in which a level indicator having the outer circumference of the same configuration is inserted, and a step-like small borehole containing an electric wire for taking out a signal to the outside is formed through the radiation shielding portion of the level indicator. The step-like configuration of the penetrating portion can decrease with ease the radiation intensity passing through the penetrating portion.

3307 NEUTRON ABSORBER DROPPING DEVICE FOR NUCLEAR REACTORS.

Giuggio, N.; Noyes, R.C. (to Combustion Engineering, Inc., Windsor, CT (USA); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,825,438/A/. 4 Jan 1979. 15p. (In German).
The device serves to release and restore a great number of neutron absorber balls (poison spheres from stainless steel) within a safety device with scram for a liquid metal reactor. Vertically coupled trap doors rest on the shoulders of a cylindrical trip gate valve within a safety shaft. The system release parameter is the coolant temperature. The trap doors form a bottom for the spheres. They are operated by tripping the gate valve and releasing the spheres that drop into the range of the safety shaft which is located in the reactor core. For restoring there is used a platform being also arranged in the safety shaft and by means of which the spheres can be lifted above the trap doors.

3308 LOW CYCLE FATIGUE BEHAVIOR OF TYPES 304 AND 316 STAINLESS STEEL TESTED IN SODIUM AT 550°C.

Zaman, G.J.; Smith, D.L. (Argonne National Lab., IL). Nucl. Technol.; 42: No. 1, 82-89 (Jan 1979).
Fatigue tests were conducted in sodium to investigate the influence of an elevated temperature sodium environment on the low cycle fatigue behavior of Types 304 and 316 stainless steel. The fatigue tests were conducted at 550°C in sodium of controlled purity, namely, 1 ppm oxygen and 0.3 ppm carbon, at a strain rate of $4 \times 10^{-3} \text{ s}^{-1}$. The fatigue life of annealed Type 316 stainless steel was substantially greater when tested in sodium than when tested in air, whereas the test environment had little influence on the fatigue life of Type 304 stainless steel. The effects of long-term sodium preexposure were also investigated. An 18-Ms (5000-h) preexposure to sodium produced little effect on the fatigue life of Type 316 stainless steel tested in sodium. However, a similar sodium exposure had a significant effect on the fatigue behavior of Type 304 stainless steel.

3309 SUMMARY OF THE ALARA NATURE OF DESIGN

EFFLUENTS FROM THE CLINCH RIVER BREEDER REACTOR PLANT. Ferguson, K.L. (Westinghouse Electric Corp., Monroeville, PA). Nucl. Technol.; 42: No. 1, 5-12 (Jan 1979).

Radioactive effluents and concentrations expected from the Clinch River Breeder Reactor (CRBR) during normal operation have been assessed. Developed were the individual off-site dose values as well as population doses within an 80-km (50-mile) radius of the proposed facility. Comparison has been with the as low as reasonably achievable (ALARA) exposure objective as in 10CFR50, Appendix I, of the Code of Federal Regulations. All estimated releases are well within the concentration limits found in 10CFR20. The calculated doses to individuals are consistently lower than values of 10CFR50, Appendix I. Population exposures are a very small fraction of that due to background radiation. Exposures due to CRBR effluents are also in conformity with cost-benefit arguments of the ALARA objective. Measures are in place to follow the design progression and operation of the facility to retain the ALARA aspect of the doses associated with the effluent releases.

3310 DYNAMIC SHAPE FACTORS FOR LMFBR MIXED-OXIDE FUEL AGGREGATES. Allen, M.D.; Moss, O.R.; Briant, J.K. (Battelle Pacific Northwest Labs., Richland, WA (USA)). J. Aerosol Sci.; 10: No. 1, 43-48 (Jan 1979).

Dynamic shape factors for aggregates of LMFBR mixed-oxide fuels have been measured with a LAPS spiral centrifuge. The aerosol was generated by pulsing a focused laser beam onto the surface of a LMFBR fuel pellet composed of (Pusub(0.2)Usub(0.8))Osub(1.96). The measured values of the dynamic shape factor depend on the aerodynamic diameter.

3311 EFFECTS OF ENERGY LOSSES OF ARBITRARY MAGNITUDE ON FUEL QUALITY, VAPOR DENSITY, AND BUBBLE DIAMETER IN AN IDEALIZED EXPANSION OF MIXTURES OF UO₂ AND SODIUM. Bertini, H.W.; White, J.S. (Oak Ridge National Lab., TN). Nucl. Sci. Eng.; 69: No. 1, 85-94 (Jan 1979).

Idealized equilibrium expansion calculations were made to determine the characteristics of UO₂ vapor bubbles produced under sodium in the capacitor discharge vaporization experiments at the Oak Ridge National Laboratory. These characteristics are the quality of the vaporized UO₂, its vapor density, and the bubble diameter. A parametric study was made over the range of initial and final states expected to occur in the experiments to determine the effects of losing arbitrary amounts of energy and of vaporizing arbitrary amounts of sodium. The results are sufficiently general to be of use in estimating bubble characteristics for a hypothetical core disruptive accident in a liquid metal-cooled fast breeder reactor.

3312 NUCLEX 78: BROAD OFFER - RESTRAINED INTEREST. REPORT ON THE 5TH INTERNATIONAL TRADE EXHIBITION AND EXPERT MEETING FOR THE NUCLEAR TECHNOLOGICAL INDUSTRY. Kalkoffen, F. (Babcock - Brown Boveri Reaktor G.m.b.H., Mannheim (Germany, F.R.)); Vollmer, H. (Hochtemperatur-Reaktorbau G.m.b.H., Mannheim (Germany, F.R.)); Morelle, J.M. (Internationale Natrium-Brutreaktor-Bau G.m.b.H., Bergisch Gladbach (Germany, F.R.)); Lurf, G. (Nuklear-Chemie und -Metallurgie G.m.b.H. (NUKEM), Hanau (Germany, F.R.)); Hoegl, A. (Frieske und Hoepfner G.m.b.H., Erlangen (Germany, F.R.)). Atomwirtsch., Atomtech.; 24: No. 1, 41-42 (Jan 1979). (In German).

From 5. international fair and technical meetings of the nuclear industries; Basel, Switzerland (3 - 7 Oct 1978).

The world's nuclear technology met in Basle between October 3 and 7, 1979 on the occasion of the 5th nuclex fair. Despite the obstacles in the way of the further expansion of nuclear programs in some countries, especially in the Federal Republic of Germany, more exhibitors were represented at this fair than had been at the last nuclex in 1975. Again, technical meetings and panel discussions accompanied the exhibition. Technical and scientific correspondents of 'atomwirtschaft' attended exhibitions and meetings and have recorded their impressions in a number of trend reports. The comprehensive review of nuclex '78 starts with some general criticism of the event and continues in five separate reports: LWR nuclear power plants; high temperature reactors; LMFBR's; nuclear fuel cycle; radiation measurement; radiological protection measurement.

3313 PURIFICATION OF SODIUM COOLANT BY MEANS OF COLD AND HOT TRAPS AND PURIFICATION OF THE COVER GAS. Subbotin, V.I.; Kozlov, F.A.; Zagorul'ko, Yu.I.; Volchkov, L.G.; Starkov, D.V.; Kononov, E.S. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.); Bagretsov, V.F.; Gryazev, V.M.; Konyashov, V.V.; Polyakov, V.I. Kernenergie; 22: No. 1, 16-24 (Jan 1979). (In German).

The basic principles as well as results of experimental studies on purification of the sodium coolant and cover gas in fast reactors are considered. Moreover, data on the reaction mechanism in the cold traps and on separation of oxygen, hydrogen, carbon, sodium-water reaction products, corrosion products, and radioactive contaminants from sodium are presented. The test results of heterophase traps for separating sodium from carbonaceous impurities are described. Finally, some problems of purifying the cover gas of radioactive contaminants are discussed.

3314 Vorschlag fuer einen Beschluss des Rates zur Festlegung eines Forschungsprogramms fuer die Europaeische Atomgemeinschaft auf dem Gebiet der Codes und Normen Schneller Reaktoren (strukturelle Integritaat von Komponenten). (PROPOSAL FOR A DECISION BY THE COUNCIL ON THE DETERMINATION OF A RESEARCH PROGRAMME OF THE EUROPEAN ATOMIC COMMUNITY IN THE FIELD OF CODES AND STANDARDS OF FAST REACTORS (STRUCTURAL INTEGRITY OF COMPONENTS)). Bonn=Bad Godesberg, Germany, F.R.; Heger (Jan 1979). 9p. (In German).

The joint programme intends to enable the industry to provide a technical basis that will make it possible to do away step by step with differences in codes and standards for specific components for liquid-metal-cooled fast breeders (LMFBR). Specific LMFBR components are non-exchangeable components which come into contact or may come into contact with sodium, sodium steam, or cover gases.

3315 KNK II - AN EXPERIMENTAL POWER STATION EQUIPPED WITH A FAST CORE. Nucl. Eng. Int.; 24: No. 208, 41-44 (Jan 1979).

The main design features of the KNK II facility, equipped with a fast reactor core in 1974 and commissioned last year, are described and main parameters are given. The facility is subject to the same licensing procedures as commercial lightwater reactor stations. Its role is to provide experience for larger breeder demonstration reactors, to test fuel

bundles, and to assist in advancing sodium technology. An outline is given of the tests undertaken since the installation of the fast core. They include irradiation and post examination of fuel elements, absorber rods and structural materials. New types of reactor instrumentation which can be tested under the special operating conditions of the KNK II are summarised. Activity in the circuit components due to activation of either sodium or corrosion products will be recorded and general behaviour of all components in contact with sodium will be monitored.

- 3316 FAST FLUX TEST FACILITY READY TO RUN. Bliss, R.J.; Corrigan, D.C.; Peckinpaugh, C.L.; Squire, A. (Westinghouse Hanford Co., Richland, WA (USA)). Nucl. Eng. Int.; 24: No. 208, 45-50 (Jan 1979).

The FFTF reactor is described and design operating conditions are given. The various phases of the Acceptance Testing Programme are outlined and experiences associated with preoperational integrated leak rate testing, testing the heating and ventilating system, and initial filling with sodium of a secondary loop are described. The programme for research and development in the FFTF reactor after February 1980 is outlined.

- 3317 ENERGY FOR THE LONG RUN: FISSION OR FUSION. Kulcinski, G.L. (Univ. of Wisconsin, Madison); Kessler, G.; Holdren, J.; Haefele, W. Am. Sci.; 67: No. 1, 78-89 (1979).

The alternatives of the most likely and controversial long-range energy sources, fusion and fast-breeder fission, are compared in several areas: potential biological and social hazards, costs of research and development, capital costs, technical complexity, and time factors. It is concluded that from biological and social hazards standpoint, fusion is preferable to fast-breeder fission reactors; however, the LMFBR has already passed on the threshold of scientific and engineering feasibility. It is pointed out that LMFBR should not be compared with short-term energy sources, e.g. coal or oil, but should be compared only with other long-term energy sources, e.g. other types of breeder reactors.

- 3318 GUIDEBOOK TO NUCLEAR REACTORS. Nero, A.V. Jr. Berkeley, CA; University of California Press (1979). 302p. The Publisher, 2223 Fulton Street, Berkeley, CA 94720 \$9.95.

The volume presents a description of the important reactor types and fuel cycles and explains how the choice of nuclear systems relates to the broader contemporary issues, such as safety, pollution, resources, and proliferation. The book provides a general introduction to power reactions, identifying basic design features, characterizing the various fuel cycles, describing power plant emissions, and discussing accident potential. The book includes an analysis of four commercial or near-commercial reactor types: PWR, BWR, PHWR, and HTGR type reactors. Advanced reactor systems are briefly discussed.

- 3319 CARBON-14 PRODUCTION IN NUCLEAR REACTORS. Davis, W. Jr. (Oak Ridge National Lab., TN). pp 151-191 of Management of low-level radioactive waste. Volume I. Carter, M.W.; Moghissi, A.A.; Kahn, B. (eds.). Elmsford, NY; Pergamon Press (1979).

From Symposium on management of low level radioactive waste; Atlanta, GA, USA (23 May 1977).

Quantities of ^{14}C that may be formed in the fuel and core structural materials of light-water-cooled reactors (LWRs), in high-temperature gas-cooled reactors (HTGRs), and in liquid-metal-cooled fast breeder reactors (LMFBRs) have been calculated by use of the ORIGEN code (Be73). Information supplied by five LWR-fuel manufacturers pertaining to nitride nitrogen and gaseous nitrogen in their fuels and fuel-rod void spaces was used in these calculations. Average nitride nitrogen values range from 3 to 50 ppm (by weight) in LWR fuels, whereas gaseous nitrogen in one case is equivalent to an additional 10 to 16 ppm. Nitride nitrogen concentrations in fast-flux test facility (FFTF) fuels are 10 to 20 ppm. The principal reactions that produce ^{14}C involve ^{14}N , ^{17}O , and (in the HTGR) ^{13}C . Reference reactor burnups are 27,500 MWD per metric ton of uranium (MTU) for boiling water reactors (BWRs), 33,000 MWD for pressurized water reactors (PWRs), about 95,000 MWD per metric ton of heavy metal (MTHM) for HTGRs, and 24,800 MWD/MTHM for an LMFBR with nuclear parameters that pertain to the Clinch River Breeder Reactor.

- 3320 FFTF GAS PROCESSING SYSTEMS.

Halverson, T.G. (Westinghouse Hanford Co., Richland, WA). pp 439-451 of Management of low-level radioactive waste. Volume I. Carter, M.W.; Moghissi, A.A.; Kahn, B. (eds.). Elmsford, NY; Pergamon Press (1979).

From Symposium on management of low level radioactive waste; Atlanta, GA, USA (23 May 1977).

The design and operation of the two radioactive gas processing systems at the Fast Flux Test Facility (FFTF) exemplify the concept that will be used in the first generation of Liquid Metal Fast Breeder Reactors (LMFBR's). The two systems, the Radioactive Argon Processing System (RAPS) and the Cell Atmosphere Processing System (CAPS), process the argon and nitrogen used in the FFTF for cover gas on liquid metal systems and as inert atmospheres in steel lined cells housing sodium equipment. The RAPS specifically processes the argon cover gas from the reactor coolant system, providing for decontamination and eventual reuse. The CAPS processes radioactive gases from inerted cells and other liquid metal cover gas systems, providing for decontamination and ultimate discharge to the atmosphere.

- 3321 ASME publication 79-PVP-25 for meeting June 25-29, 1979, 12p. (INELASTIC ANALYSIS OF THE UPPER INTERNALS STRUCTURE FOR THE CLINCH RIVER BREEDER REACTOR PLANT). Falk, P.T.; Kosanich, M. Sr. Contract EW-76-C-15-2395. New York, NY; American Society of Mechanical Engineers (1979). 32p. (CONF-790615-3). American Society of Mechanical Engineers, 345 East 47th Street, New York, NY.

From 3. US national congress on pressure vessels and piping; San Francisco, CA, USA (25 Jun 1979).

Inelastic analysis is required for the Upper Internals Structure (UIS) of the Clinch River Breeder Reactor due to its operation in an elevated temperature environment where creep effects are significant. The UIS is a permanent reactor structure and is thus designed to survive in this environment for a 30-year design life. The rigorous inelastic finite element analyses of the critical lower plate ligament are presented. The interaction between the gross model, plate model and ligament model is also described. The analysis results are evaluated temperature design codes required to demonstrate the structural adequacy of the

design. The transient conditions, the environmental effects on mechanical properties, elevated temperature analysis guidelines, and the simplified analysis techniques used in determining the critical areas are described.

- 3322 3RD PART LICENCE FOR SNR-300, KALKAR (NO. 7;3 SNR). GRS Kurz-Inf., Reihe B; No. 2, 1-4(1979). (In German).

The 3rd part licence deals with the erection of a number of reactor components. Immediate realisation of this project is advisable in view of the particular public interest in the completion of the construction work.

- 3323 INSTRUMENT CALIBRATION PROGRAM FOR STARTUP OF FFTF. Swanson, C.D. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc., Suppl.; 32: No. 1, 36-37(1979).
FFTF REACTOR; REACTOR INSTRUMENTATION; CALIBRATION

- 3324 TREATMENT OF DEPENDENCIES IN RELIABILITY ANALYSES. Elerath, J.G.; Ingram, G.E. (GE, Sunnyvale, Calif). Proc., Annu. Reliab. Maintain. Symp.; 99-103(1979).

This paper presents a method for the evaluation of dependencies in discrete state reliability models. The theoretical basis for evaluation and the computer code PROBCALC which makes this method tractable for any system are presented. Examples of model dependencies are taken from analyses of Shutdown Heat Removal Systems in loop type Breeder Reactors. The general technique and the use of PROBCALC are both illustrated by application to an example Shutdown Heat Removal System.

- 3325 POTENTIAL OF ADVANCED GENERATION TECHNOLOGIES FOR NORTHEAST UTILITIES SYSTEM. Platts, J.E. New York, NY; Institute of Electrical and Electronics Engineers, Inc. (1979). 9p. (CONF-790214--).

From 1979 winter power meeting; New York, NY, USA (4 Feb 1979).

A study conducted by the Long Range System Planning Section of Northeast Utilities (NU) evaluated the potential application of eight advanced generation technologies for the development of the NU generation system in the next 30 years. The technologies studied were coal gasifier-combined cycle, coal gasifier-fuel cell, liquid metal fast breeder reactor, gas turbine-high temperature gas reactor, fuel cells, advanced combined cycle, batteries and advanced gas turbines. The study determined the economic competitiveness of these technologies compared to present options using busbar costs, and economic generation expansions over a 20 year period. Additional system benefits potentially offered by the advanced technologies were also identified. The study recommended that NU's future generation research should focus on the gasifier-combined cycle, gasifier-fuel cell, and gas turbine HTGR. The study results point to future potential for utility owned cogeneration with these technologies.

- 3326 PRIMARY AEROSOL PARTICLE-SIZE DISTRIBUTIONS FROM HOMOGENEOUS NUCLEATION CONDENSATION AND PARTICLE GROWTH. Kennedy, M.F. Charlottesville, VA; University of Virginia (1979). 121p. University Microfilms Under No. 79-16,275.
Thesis (Ph. D.).

The process of forming aerosols by homogeneous nucleation condensation followed by

condensation growth was studied as a potential source of submicrometre aerosols in LMFBR accidents. A model was developed for calculating particle-size distributions following nucleation and growth. This model was applied to experiments by ORNL in which UD_2 fuel pellets were partially vaporized in an argon atmosphere and the resulting primary particle-size distribution was measured. The low range of particle sizes observed in the ORNL tests could be reproduced by the model by assuming large mixing rates between the fuel vapor and the argon, thus indicating that homogeneous nucleation condensation was likely the source of the small particles observed.

- 3327 INTEGRATED QUALITY STATUS AND INVENTORY TRACKING SYSTEM FOR FAST FLUX TEST FACILITY DRIVER FUEL PINS. Gottschalk, G.P. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc., Suppl.; 33: No. 7, 26-27(1979).
FFTF REACTOR; FUEL PINS; FUEL ASSEMBLIES; REACTOR INSTRUMENTATION; FUEL MANAGEMENT

- 3328 RADIATION EFFECTS IN STRUCTURAL MATERIALS. Stiegler, J.O.; Mansur, L.K. (Oak Ridge National Lab., TN). Annu. Rev. Mater. Sci.; 9: 405-454(1979).

When energetic particles impinge on a solid, atoms are displaced from their lattice positions. Single or clustered vacant atomic sites are produced and the displaced atoms come to rest in interstitial configurations. The bombarding particles may also induce nuclear transmutations. The initial disturbance of the material is termed radiation damage. At finite temperatures or under continued irradiation the defects migrate; some annihilate and others cluster, both among themselves and with solute atoms. The resulting microstructural changes in the material are more stable than the initial damage and lead to changes in properties termed radiation effects. In particular, changes in dimensions, composition on a microscale, and mechanical properties may be extreme.

- 3329 INFLUENCE OF CORE ASSEMBLY REFUELING REQUIREMENTS ON LMFBR CORE SYSTEM DESIGN. Fox, J.N. (GE, Sunnyvale, Calif). Am. Soc. Mech. Eng., [Pap.]; No. 79-PVP-33, 8(1979).
Liquid metal fast breeder reactor (LMFBR) core assemblies are exposed to an operational environment which induces permanent distortions in their main structural members. These distortions have a substantial impact on core assembly refueling since the distortions are large compared to the available spaces. Core assembly refueling requirements demand that refueling be accomplished without damage to adjacent core components or to the refueling equipment. This paper describes the core assembly refueling requirements and the design procedures used to demonstrate compliance with the requirements. This paper also provides an assessment relative to the influence of these requirements on LMFBR core system design.

- 3330 DESIGN CONSIDERATIONS IN LIQUID METAL FAST BREEDER REACTOR UPPER INTERNALS STRUCTURES. Wiltshire, F.R.; Kraus, S. (GE, Sunnyvale, Calif). Am. Soc. Mech. Eng., [Pap.]; No. 79-PVP-34, 7(1979).

Various functional requirements in Liquid Metal Fast Breeder Reactors (LMFBR's) necessitate that permanently installed structures be located directly above the reactor core. In present designs these structures are supported from the reactor closure and extend downward, through the

reactor outlet plenum, to the top of the core assemblies. They are, hence, subject to the severe thermal and hydraulic environment that is typical of the core exit plane. This paper reviews these functional requirements and operating conditions and discusses the impact that these factors have on the design of reactor Upper Internals Structures (UIS). The current design approaches to solving the problems are presented. Recommendations are proposed for future work to provide a better understanding of the problems and to develop design solution to these problems.

- 3331 NUMERICAL EVALUATION OF AN INELASTIC PIPING ELBOW ELEMENT. Scheller, J.D.; Mallett, R.H. (McDonnell Douglas Autom Co, St. Louis, Mo). Am. Soc. Mech. Eng., [Pap.]; No. 79-PVP-41, 15(1979).

The PIRAX computer program embodies a simplified model for predicting inelastic behavior of pipe elbows at elevated temperature. This model is evaluated for use in inelastic flexibility analysis of LMFBR (Liquid Metal Fast Breeder Reactor) piping by comparing results obtained using PIRAX with results obtained from more complex analytical models and tests. Isolated piping elbows are examined under in-plane and out-of-plane loading. Plasticity, thermal creep under load-controlled loading, and relaxation by thermal creep under deformation controlled loading are considered. The load versus deformation behavior obtained using the simplified model for piping elbows is judged to be a usable approximation. Based upon this evaluation, it is concluded that the inelastic flexibility analysis capability of the PIRAX computer program should be useful for detecting elastic followup in LMFBR piping systems and for classifying forces induced in LMFBR piping systems into load-controlled and deformation-controlled categories. 22 refs.

- 3332 ELEVATED TEMPERATURE, CYCLIC LOADINGS AND IRRADIATION EFFECTS ON FATIGUE CRACK OF LMFBR PRESSURE VESSELS. Ma, B.M. (Iowa State Univ, Ames). Am. Soc. Mech. Eng., [Pap.]; No. 79-PVP-59, 1-10(1979).

The elevated temperature, cyclic loadings and irradiation effects on the fatigue crack of the base and weld materials of an LMFBR pressure vessel are analyzed on the basis of experimental data. The power function law and the exponential function law of stress intensity factor range, ΔK , and effective stress intensity factor, $K_{sub\ eff}$, for the fatigue crack growth rate of the vessel materials (stainless steel Types 304 and 316 and nickel alloy Inconel 600) are, respectively, proposed and the equations are defined and derived. Based on these equations, relatively complete published and unpublished experimental data of fatigue crack growth rate for the vessel materials are reorganized, analyzed and plotted in the figures shown. At given temperature and fast neutron fluence, the high cyclic loading or low cyclic frequency can produce a large fatigue crack propagation rate in the pressure vessel.

- 3333 PRELIMINARY ANALYSIS AND SCREENING CRITERIA FOR ELEVATED TEMPERATURE PIPING. Mimani, H.M.; Naugle, F.V. Contract EY-76-C-03-0824-021. New York, NY: American Society of Mechanical Engineers (1979). 7p. (CONF-790615-28). \$3.00.

From 3. US national congress on pressure vessels and piping; San Francisco, CA, USA (25 Jun 1979).

The complexities of the ASME Class 1 nuclear elevated temperature structural design criteria

precludes the possibility of performing a complete Code Case 1592 analysis for all of the alternative designs normally investigated in the course of the design of a Liquid Metal Fast Breeder Reactor (LMFBR) high-temperature piping system. To enable the piping design to proceed on an orderly controlled basis, the piping design analysis is separated into the following phases: (a) screening analysis to determine pipe routing feasibility, (b) preliminary analysis to determine pipe support configuration, and (c) final design analysis (including transients) to demonstrate Code compliance. The application of the first phases of this procedure is demonstrated by a summary of the analyses performed in the development of the preliminary design configuration for the primary system of a commercialized LMFBR.

- 3334 TRIAL APPLICATION OF THE DRAFT STRUCTURAL DESIGN CRITERIA FOR BREEDER REACTOR CORE COMPONENTS TO A TYPICAL BLANKET ASSEMBLY DUCT; SUP 1. Sazawal, V.K. (Westinghouse Electr Corp, Madison, Pa). Am. Soc. Mech. Eng., [Pap.]; No. 79-PVP-27, 12(1979).

This paper presents the results of a comprehensive trial application of the draft structural design criteria for highly irradiated core components to a typical blanket assembly duct. The development of the criteria (proposed RDT F 9-7) is sponsored by the United States Department of Energy. The activity focuses on evaluation of the criteria using a typical breeder reactor blanket assembly as a test case and is not intended for judging the adequacy of the component design. Pertinent design rules from the draft criteria are applied to the component to determine: difficulties, if any, that may arise in interpretation and application of the criteria, and adequacy of the criteria rules and analysis methods to address various potential failure mechanisms expected in the reactor core during service. Based on the trial application, various recommendations are made to promote further development of the criteria. 15 refs.

- 3335 LOCAL PERTURBATION OF NUCLEAR REACTOR COOLING. A REMARKABLE EXPERIENCE. Vanmassenhove, G. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)); Kramer, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). Consensus; No. 1, 17-31(1979). (In French).

Fuel element behaviour under local off normal cooling conditions and possible pin-to-pin failure propagation are of special interest in the safety analysis of fast reactors. In the experiments (loops Mol7C), the aim is to demonstrate that failure propagation does not occur and that total cooling of the pin bundle is not endangered even if the pins in a large part of the cross section of the bundle have failed.

- 3336 Grundsätze der Instrumentierung von flüssigmetallgekühlten schnellen Reaktoren. Entwurf. (PRINCIPLES OF INSTRUMENTATION OF LIQUID-METAL COOLED FAST REACTORS. DRAFT). Berlin, Germany, F.R.; Beuth (1979). 5p. (In German).

Instrumentation for measuring neutron flux density, coolant and reactor vessel, temperatures, for coolant monitoring, for monitoring of the protective gas systems and for reactor control.

- 3337 POST-SCRAM LMFBR HEAT TRANSPORT SYSTEM DYNAMICS. Brukx, J.F.L.M.; Latzko, D.G.H. Am. Soc. Mech. Eng., [Pap.]; No. 79-HT-8,

12(1979).

Using hybrid computer simulation, the paper evaluates alternative feed water control schemes for a loop-type liquid-metal-cooled fast breeder reactor heat transport system under post-scrum decay heat removal conditions. The results show the need for a secondary sodium flow feedforward signal in order to limit temperature transients. In addition, the paper deals with transition from forced to natural circulation of the primary and secondary sodium flows. 9 refs.

- 3338 FATIGUE CRACK GROWTH BEHAVIOR OF STAINLESS STEEL TYPE 316 PLATE AND 16-8-2 WELDMENTS IN AIR AND HIGH-CARBON LIQUID SODIUM. Yuen, J.L.; Copeland, J.F. (GE, San Jose, Calif.). Am. Soc. Mech. Eng., [Pap.]; No. 79-PVP-105, 13(1979).

The fatigue crack growth rates of Type 316 stainless steel and 16-8-2 gas tungsten arc welds were measured in air and high-carbon liquid sodium at 510 C (950 F) for secondary sodium loop Liquid Metal Fast Breeder Reactor applications. Loading variables such as low frequency, load ratio, and hold-time at peak load were studied. Supporting studies show that carburization of crack faces occurs prior to any crack tip carburization for the crack sizes and load frequencies of this study. Thus, a carburization effect on da/dN is not shown for relatively short-term tests. Low stress intensity, ultra-low frequency tests to simulate service would be required to show this effect. 16 refs.

- 3339 DEPRESSURIZATION OF INTERNALLY HEATED BOILING POOLS. Henry, R.E.; Smith, J.L.; Weber, J.T. (Argonne Natl Lab, Ill). Am. Soc. Mech. Eng., [Pap.]; No. 79-HT-101, 8(1979).

In assessing the behavior of molten fuel during hypothetical accident conditions, it is necessary to evaluate the behavior of boiling pools with internal heat generation under pressurized and blowdown (depressurization) conditions. In this experimental investigation, depressurization experiments were conducted with water in a transparent lucite vessel with various initial void fractions. Experiments were also carried out with bulk nucleation sites "seeded" in the liquid phase and with no such "seeding". The experiments show that with bulk nucleation sites the mixture maintains a homogeneous character during blowdown and essentially the entire liquid mass is vented during the depressurization. Without bulk nucleation sites, there is significant drift of vapor through the liquid phase and considerably less liquid is vented during the transient. 5 refs.

- 3340 IMPACT OF CLINCH RIVER BREEDER REACTOR PLANT ON BREEDER RESEARCH AND DEVELOPMENT PROGRAMS. Elias, D. (Clinch River React Plant Proj, Oak Ridge, Tenn). Am. Soc. Mech. Eng., [Pap.]; No. 79-JPGC-NE-5, 1-16(1979).

The Clinch River Breeder Reactor Plant (CRBRP), the first U.S. large-scale breeder reactor demonstration plant (375 MWe), is being developed jointly by the Government and industry to demonstrate the potential of a large-scale Liquid Metal Fast Breeder Reactor (LMFBR) as a future source of economic electric power in a utility environment. This paper summarizes CRBRP specific and generic research and development programs in support of the breeder program. General areas highlighted include fuels and materials, components, systems and analytical, and safety development programs. It provides information on the way

the development programs have been integrated with the CRBRP and managed to maximize the benefits to the United States. Examples relate development programs with large hardware being designed, fabricated, and delivered for operation. The adverse effects of not proceeding with this demonstration plant on development programs and on the ability to obtain information from foreign sources are discussed.

- 3341 COMPARATIVE ASSESSMENT OF THE LMFBR AND ADVANCED CONVERTER FUEL CYCLES. Heising-Goodman, C.D.; Saragossi, I.; Sharafi, P. (MIT, Cambridge, Mass). Am. Soc. Mech. Eng., [Pap.]; No. 79-JPGC-NE-3, 1-8(1979).

Three advanced reactor technologies and their fuel cycles were compared with respect to their relative economic potential and diversion resistance. These systems include the LWR once-through cycle with and without extended burnup, the LWR and LMFBR on uranium/plutonium recycle with and without CIVEX reprocessing methods and lastly, advanced converters in symbiosis with breeders on a denatured thorium cycle. To quantify relative diversion resistance, a method based on multi-attribute decision theory was applied. Economic potential was measured on the basis of a simple mills/kWh fuel cycle cost model. Preliminary results indicate that the LWR-LMFBR (U/Pu) cycle can be made as diversion resistant as the LWR once-through cycle if CIVEX-like methods are utilized. The denatured thorium cycle shows no obvious diversion resistance advantage and exhibits some economic penalty when compared to the other alternatives. The economic calculations show that the choice between cycles is less likely to be affected by direct economic considerations than by institutional factors such as fuel cycle implementability on an industry-wide basis, acceptability to U.S. regulators and the final prospects for their adoption by U.S. utilities. 15 refs.

- 3342 CURRENT PRACTICES FOR ULTRASONIC AND RADIOGRAPHIC EXAMINATION OF TUBES, TUBE PLATES AND TUBE-PLATE WELDS OF TUBE BUNDLES IN HEAT EXCHANGERS. CHAPTER 3. Hudgell, R.J.; Tickle, H. (UKAEA, Risley, Nuclear Power Development Establishment). pp 7-100 of Developments in pressure vessel technology - 2. Inspection and testing. Nichols, R.W. (ed.). London; Applied Science (1979).

The chapter describes the ultrasonic and radiographic inspection procedures that are applied to heat exchanger tube bundles. The inspection process starts with the ultrasonic examination of the tubes and tube plates during manufacture, followed by radiography of the tube-to-tube-plate welds during fabrication of the tube bundle. Ultrasonic methods are explained for welds which are amenable to this type of inspection. For the in-service inspection of tube bundles the chapter relates the authors' experiences on the ultrasonic inspection of tubes and tube plates in the Prototype Fast Breeder Reactor at Dounreay. At the end of the chapter some comments are made about future ultrasonic and radiographic developments for tube bundles.

- 3343 CLINCH RIVER BREEDER REACTOR PLANT: A BUILDING BLOCK IN NUCLEAR TECHNOLOGY. McCormack, M. (United States House of Representatives, Washington, DC). Electr. Perspect.; No. 79/4, 22-28(1979).

Interest in breeder reactors dates from the Manhattan Project to the present effort to build the Clinch River Liquid Metal Fast Breeder Reactor (LMFBR) demonstration plant.

Seven breeder-type reactors which were built during this time are described and their technological progress assessed. The Clinch River Breeder Reactor Project (CRBRP) has been designed to demonstrate that it can be licensed, can operate on a large power grid, and can provide industry with important experience. As the next logical step in LMFBR development, the project has suffered repeated cancellation efforts with only minor modifications to its schedule. Controversies have developed over the timing of a large-scale demonstration plant, the risks of proliferation, economics, and other problems. Among the innovative developments adopted for the CRBRP is a higher thermal efficiency potential, the type of development which Senator McCormack feels justifies continuing the project. He argues that the nuclear power program can and should be revitalized by continuing the CRBRP.

- 3344 COMPUTER ANALYSIS AND CONTROL FOR NONDESTRUCTIVE GAMMA SCANNING OF LMFBR FUEL PINS. Barnes, B.K.; Phillips, J.R.; Waterbury, G.R.; Quintana, J.N.; Murray, A.S.; Netuschil, J.R. (Univ. of California, Los Alamos, NM). pp 801-810 of Computers in activation analysis and gamma-ray spectroscopy. Carpenter, B.S.; D'Agostino, M.D.; Yule, H.P. (eds.). Washington, DC; Department of Energy (1979).

From Conference on computers in activation analysis and gamma-ray spectroscopy; Mayaguez, Puerto Rico (30 Apr 1978).

A program to modernize existing computer control and data analysis in a precision gamma-scanning system is described. This involves interfacing a new computer-based multichannel analyzer to existing equipment through a CAMAC system with an intelligent crate controller and writing software for scan control and data reduction. Both hardware and software requirements are discussed. 6 figures.

- 3345 NUCLEAR SAFETY IN FRANCE. Tanguy, P. (CEA Centre d'Etudes Nucleaires de Fontenay-aux-Roses, 92 (France). Inst. de Protection et de Surete Nucleaire). Ann. Nucl. Energy; 6: No. 5, 299-304(1979).

A brief description of the main safety aspects of the French nuclear energy programme and of the general safety organization is followed by a discussion on the current thinking in CEA on some important safety issues. As far as methodology is concerned, the use of probabilistic analysis in the licensing procedure is being extensively developed. Reactor safety research is aimed at a better knowledge of the safety margins involved in the present designs of both PWRs and LMFBRs. A greater emphasis should be put during the next years in the safety of the nuclear fuel cycle installations, including waste disposals. Finally, it is suggested that further international cooperation in the field of nuclear safety should be developed in order to ensure for all countries the very high safety level which has been achieved up till now.

- 3346 THORIUM FUEL CYCLES. Yang, C.S. Berkeley, CA; Univ. of California (1979). 184p. University Microfilms Order No. 80-00,578.

Thesis (Ph. D.).

Thorium fuel cycles for light-water reactors, heavy-water CANDU-type reactors and high-temperature gas-cooled reactors have been studied in this thesis. Fissile makeup consists of enriched ^{235}U or plutonium from uranium-

fueled reactors. Uranium resource requirement for each of the thorium fuel cycles has been calculated and compared with the uranium requirement of the fuel cycles of the corresponding uranium-fueled reference designs. Using thorium in the existing light-water reactors is the easiest way of introducing thorium fuel cycles in this country. The uranium ore requirement over a 30-year reactor lifetime, for the fully-enriched-uranium-thorium fuel cycle is only 17% less than that for the uranium fuel cycle, with uranium and plutonium recycle. Among all the fuel cycles studied in this thesis, the 93.5% ^{235}U -Th CANDU fuel cycle needs the least uranium resources. Self sustaining thorium breeding is also a possibility, with only small changes to the thorium-fueled CANDU reactor operating conditions. Therefore, this fuel cycle could be a logical choice if there were a serious delay in the commercialization of liquid-metal fast breeder reactor.

- 3347 LOW HEAT FLUX SODIUM VOIDING USING THE DRIFT FLUX FORMULATION. Senglaub, M.E. Chicago, IL; Northwestern Univ. (1979). 168p. University Microfilms Order No. 79-27,447.

Thesis (Ph. D.).

The drift flux formulation has been used to study the conditions of low-heat flux boiling in sodium-cooled fast-reactor systems. Models and numerical methods have been developed in an effort to study such events as the loss of heat sink type of transient in the next generation of fast reactors. A detailed derivation of the four-equation drift flux formulation is given with the required simplifying assumptions. The additional assumptions and physical arguments that have been used to arrive at the final formulation are presented. Along with the mathematical formulations, the derivation of the solution numerics, and the stability analysis of the procedure used have been carried out. To test the proposed numerics a number of cases were calculated for comparison with the limited number of existing experiments. In addition, a number of numerical experiments were performed to examine the capabilities of the solution techniques in regimes for which no experimental data exist. Finally, the applicability range has been established within physical constraints and the possible approaches for extending this range of applicability have been considered.

- 3348 (RISLEY-Trans--4087) STATE OF WORK ON FAST REACTORS IN THE USSR UP TO MARCH 1979. Inyutin, E.I.; Krasnoyarov, N.V. 1979. Translation of paper presented at 12th annual IWGFR meeting, Vienna, Austria, March 27-30, 1979 12p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Information is presented concerning the BR-10 reactor; BOR-60; BN-350; BN-600 and BN-1600; and research activities.

- 3349 MICROSTRUCTURAL OBJECTIVES FOR HIGH-TEMPERATURE ALLOYS IN ADVANCED ENERGY SYSTEMS. Sims, C.T. (GE, Schenectady, NY). Am. Soc. Test. Mater., Spec. Tech. Publ.; No. 672, 480-513(1979).

From ASTM meeting on microstructural control; Houston, TX, USA (3 Apr 1978).

This paper reviews six advanced systems in a broad fashion helpful in orientation of engineering and technical personnel not directly involved in such developments. These are coal utilization systems including coal liquefaction, coal gasification and pressurized fluid beds, nuclear systems including liquid metal fast breeder reactors (LMFBR), and high-

temperature gas reactors (HTGR). In concert with these systems, yet cross-functioning, superalloy technologies are discussed for combined-cycle electrical plants and high-temperature heat exchangers. Then potential applications and problems for superalloys in the systems, are identified, drawing from this base examples that might typically be applied, utilizing these to discuss ideal microstructures one may find. 18 refs.

- 3350 NUCLEAR OXIDE FUEL ANALYSIS. Timofeev, G.A.; Gabeskiria, V.Ya.; Simakin, G.A.; Mishenev, V.B.; Bezz, A.S.; Rykov, A.G. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). J. Radioanal. Chem.; 51: No. 2, 377-383(1979).

From 9. Radiochemical conference; Piestany, Czechoslovakia (11 - 16 Sep 1978).

As a rule the analysis of nuclear oxide fuel includes the determination of uranium, plutonium, their isotopic composition, cation impurities, carbon, nitrogen, chlorine, fluorine, oxygen coefficient. In this paper different methods are discussed for the analysis of unirradiated uranium and plutonium oxide fuels used in the laboratories of the Analytical Chemistry Department: coulometry, emission analysis, chromatography, X-ray analysis. Much consideration is being given to the analysis of uranium and plutonium oxide samples and uranium-plutonium mixed fuels irradiated in the BOR-60 using mass-spectrometric (isotope dilution method) and radiometric techniques. The results of uranium and plutonium determination by these methods are compared. The main analytical characteristics of the methods are given.

- 3351 BREEDER REACTORS--WHAT ARE THE PROSPECTS.. Behnke, W.B. (Common Edison Co, Ill). Energy Technol. (Wash., D.C.); 6: 851-856(1979).

A discussion is presented which covers the following five points: first, the inseparable link between the breeder reactor and further expansion of nuclear electric power generation; second, the present stagnation of nuclear power in the United States; third, the Administration's energy policy and its significance in light of current world events; fourth, a suggestion on how we can get nuclear power development going forward again; and fifth, completion of the Clinch River Breeder Reactor and why it is the right choice for the next step in our breeder demonstration program. 7 refs.

- 3352 CONTROL OF UNIRRADIATED FUEL IN BOR-60 FAST REACTOR FUEL ASSEMBLIES BY NEUTRON RADIATION. Gryazev, V.M.; Gadzhiev, G.I.; Leshchenko, Yu.I.; Gorobets, A.K.; Semenov, A.L. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). pp 521-530 of Nuclear safeguards technology 1978. Vol. 1. Vienna, Austria; International Atomic Energy Agency (1979). (In Russian)

From IAEA symposium on nuclear material safeguards; Vienna, Austria (2 Oct 1978).

Experimental studies have been made of the possibility of monitoring the characteristics of unirradiated fuel assemblies for the BOR-60 reactor by means of the natural neutron radiation and the delayed-neutron emission initiated by a ^{252}Cf source. The experimental apparatus uses a neutron detector with a high efficiency (approximately 13% for the neutrons from the californium source) and a pneumatic transport system with a approximately 5×10^7 n/s californium source. A coincidence technique is employed to separate the spontaneous-fission

and the (α, n) -reaction neutrons in the natural neutron radiation of the fuel. The possibility of monitoring the fuel characteristics by means of delayed neutron emission has been tested by studying the sensitivity of the measurement results to the removal of fuel from the assemblies. For a fuel with a known enrichment, the measured values seem to be linearly related to the amount of ^{235}U over a limited range of fuel contents. An analysis of the results has shown that an estimate of the amount of ^{235}U in a fuel assembly can be obtained to an accuracy of better than 2% (2b) in a measurement time of 30 minutes. For fuel where the enrichment is unknown, use is made of the dependence of the measured values on the spectrum of the neutrons irradiating the assembly. The results for unirradiated fuel in BOR-60 assemblies show good agreement with the rated ^{235}U content. In connection with the use of this apparatus in safeguards systems, attention has been given to the possibility of falsification of the results and to ways of detecting this.

- 3353 BASIC PRINCIPLES OF ACCOUNTING AND CONTROL OF NUCLEAR MATERIALS IN THE BOR-60 EXPERIMENTAL FAST REACTOR. Gryazev, V.M.; Gadzhiev, G.I.; Alekseev, I.N. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). pp 531-543 of Nuclear safeguards technology 1978. Vol. 1. Vienna, Austria; International Atomic Energy Agency (1979). (In Russian)

From IAEA symposium on nuclear material safeguards; Vienna, Austria (2 Oct 1978).

Under a contract with the International Atomic Energy Agency, the V.I. Lenin Atomic Reactor Research Institute is currently carrying out a study of ways of organizing a nuclear materials accounting and control system for the BOR-60 fast reactor. Some results of this study are presented in the paper. The special physical and technological features of fast reactors create additional difficulties in safeguards systems and give rise to a number of new possibilities for the illicit removal of nuclear materials. These questions are discussed with reference to the BOR-60 reactor but the conclusions are probably applicable to all fast reactors. The proposed accounting and control system is based on non-destructive measurements of the amount of fissile materials in the operating fuel assemblies and screened bundles of the reactor, on the independent control of the principal facility parameters (a list of which is given) and on an automated information collection and evaluation system. Visual means of inspection can be very effective in fast reactor safeguards systems, especially for controlling storage, but they are not used with the BOR-60 reactor.

- 3354 STATUS OF THE SAFEGUARDS SYSTEM DEVELOPED FOR THE LMFB PROTOTYPE POWER PLANT SNR-300 (KKW KALKAR). Krinninger, H. (Internationale Natrium-Brutreaktor-Bau G.m.b.H., Bergisch Gladbach (Germany, F.R.)); Brueckner, C. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)); Quandt, U.; Ruppert, E. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)); Tschampel, H. (Dr. Seufert GmbH, Karlsruhe, Germany, F.R.); Hulst, P. van der (Schnell-Brueter-Kernkraftwerksgesellschaft m.b.H. (SBK), Essen (Germany, F.R.)). pp 483-506 of Nuclear safeguards technology 1978. Vol. 1. Vienna, Austria; International Atomic Energy Agency (1979).

From IAEA symposium on nuclear material safeguards; Vienna, Austria (2 Oct 1978).

In this report the features of the safeguards system developed for the LMFB

- Prototype Power Plant SNR-300 (KKW Kalkar) are described. Due to the fact that LMFBR fuel assemblies are mostly handled in a sodium or inert gas environment, visual control and counting of the assemblies containing nuclear material, as routinely used in a LWR power plant, cannot be applied for LMFBRs. Consequently, for safeguarding the Kalkar Nuclear Power Station, an automatic and continuous monitoring of the fissile material inventory is the main objective of special instrumentation, the "Inaccessible Inventory Instrumentation System" (IIIS). For a qualitative distinction between the various types of assembly the detection of the emitted neutron and gamma radiation turned out to be most adequate. Based upon this conclusion and taking into account its tasks, the IIIS is composed of an activity measurement system, a position-sensing system and a micro-computer system (MCS). The high reliability, high fraud resistance and low maintenance requirements have had an important influence on the IIIS design. Access to IIIS data is protected by special hardware and software measures. To protect the system against the results of power failures the IIIS possesses an automatic restart facility and a battery-buffered power supply for the RAM-unit of the microprocessor system used for data acquisition and evaluation.
- 3355 SUPERPHENIX IN DETAIL. CONSEQUENT EXECUTION OF CONSTRUCTION PROGRAMS FOR NUCLEAR POWER PLANTS. Weber, R. VDI (Ver. Dtsch. Ing.) Nachr.; 33: No. 35, 9(1979). (In German).
- On the banks of the Rhone river, 60 km east of Lyon, the Superphenix nuclear power plant is being built. It is the prototype of a reactor line that will make up one fourth of the nuclear capacity in France already at the turn of the century. The principles and experiences of Superphenix are reviewed, taking only the most important of the many technical details.
- 3356 HALF-TIME IN THE CONSTRUCTION OF 'SUPERPHENIX'. THE NUCLEAR POWER PLANT SERVES AS A MODEL FOR THE FUTURE FRENCH STANDARD BREEDER. Weber, R. VDI (Ver. Dtsch. Ing.) Nachr.; 33: No. 34, 4(1979). (In German).
- Only the USSR and France are consequent in their way to the breeder reactor. The 1200 MWe Superphenix started in 1976 has reacted half-time for its construction according to schedule. This fast breeder, by far the biggest in the world in the eyes of French energy experts marks the beginning of serial fabrication and thus an important step to the independence from foreign energies.
- 3357 DESCRIPTION OF THE SODIUM LOOP ML-3. de la Torre, M.; Melches, I; Lapena, J.; Martinez, T.A.; De Miguel, D.; Duran, F. Energ. Nucl. (Madrid); 23: No. 121, 273-281(1979). (In Spanish).
- The sodium loop ML-3 is described. The main objective of this facility is to obtain mechanical property data for LMFBR materials in creep and low cycle fatigue testing in flowing sodium. ML-3 includes 10 test stations for creep and two for fatigue. It is possible to operate simultaneously at three different temperature levels. The maximum operating temperature is 650 deg C at flow velocities up to 5 m/s. The ML-3 loop has been located in a manner that permits the fill/dump tank cover gas and security systems to be shared with an earlier circuit, the ML-1.
- 3358 MAGNETIC NUCLEAR CORE RESTRAINT AND CONTROL. Cooper, M.H. (to Dept. of Energy). US Patent 4,131,510. 26 Dec 1978. Filed date 21 Jun 1977. 6p.
- Disclosed is a lateral restraint and control system for a nuclear reactor core adaptable to provide an inherent decrease of core reactivity in response to abnormally high reactor coolant fluid temperatures. An electromagnet is associated with structure for radially compressing the core during normal reactor conditions. A portion of the structures forming a magnetic circuit are composed of ferromagnetic material having a curie temperature corresponding to a selected coolant fluid temperature. Upon a selected signal, or inherently upon a preselected rise in coolant temperature, the magnetic force is decreased a given amount sufficient to relieve the compression force so as to allow core radial expansion. The expanded core configuration provides a decreased reactivity, tending to shut down the nuclear reaction.
- 3359 HORIZONTAL BAFFLE FOR NUCLEAR REACTORS. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). British Patent 1,536,645/A/. 20 Dec 1978. 10p.
- A physical partition in the annulus between the core barrel and a thermal barrier of the reactor vessel of an LMFBR prevents return of flow of the hot coolant and uses thermal bowing to enhance sealing and prevents contact of the coolant with the cooler structural members in the annulus below the partition.
- 3360 DETECTION DEVICE FOR FUEL FAILURE. Mimoto, Y.; Kitagawa, H. (to Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Japanese Patent 1953-146,094/A/. 19 Dec 1978. 4p. (In Japanese).
- Purpose: To enable inspection for actual failures in the fuels of reactor core that are suspected to be failed with a high accuracy by the detection of the presence or absence of gaseous fission products leaked from failed fuel pins due to expansion under reduced pressure. Constitution: Fuel pins to be inspected in a reactor core suspected of failures are taken out of the reactor core by the drive of a fuel exchanger and then inserted in a containment pipe. Then, the lower end of the upper pipe is engaged to the upper end of the containment pipe and the pressure therein is reduced. If the fuel pins to be inspected in the containment pipe has actually failed, the gaseous fission products remained in the fuel pins are expanded under the reduced pressure and leak out of the failed fuel pins. Thereafter, the valve is opened and the gaseous products are introduced into a gaseous fission products detector and the presence of the products is checked.
- 3361 REACTOR COOLING SYSTEM. Karamori, A. (to Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Japanese Patent 1953-143,892/A/. 14 Dec 1978. 3p. (In Japanese).
- Purpose: To decrease the cover gas pressure in the reactor containment while ensuring reactor safety well by the provision of a main cycling pump to a cold leg. Method: The other end of a degassing pipeway provided at the upper portion of an intermediate heat exchanger is inserted in a degassing pot, and the degassing pot is connected by way of a pipe line with the top of the reactor containment. Upon driving, the main cycling pump is driven to circulate coolants such as liquid sodium in a coolant loop. The heat generated in a reactor

core is gradually removed by the coolant flowing passing therethrough and the coolants heated to a high temperature is heat exchanged with secondary coolants in the intermediate heat exchangers.

- 3362 GAMMA PRIME HARDENED NICKEL--IRON BASED SUPERALLOY. Korenko, M.K. (to Dept. of Energy). US Patent 4,129,462. 12 Dec 1978. Filed date 7 Apr 1977. 8p. PAT-APPL-785,640.

A low swelling, gamma prime hardened nickel--iron base superalloy useful for fast reactor duct and cladding applications is described having from about 7.0 to about 10.5 weight percent (wt%) chromium, from about 24 to about 35 wt% nickel, from about 1.7 to 2.5 wt% titanium, from about 0.3 to about 1.0 wt% aluminum, from about 2.0 to about 3.3 wt% molybdenum, from about 0.05 to about 1.0 wt% silicon, from about 0.03 to about 0.06 wt% carbon, a maximum of about 2 wt% manganese, and the balance iron.

- 3363 METHOD AND EQUIPMENT FOR THE PERFORMANCE REDUCTION OF A NUCLEAR REACTOR DURING TEMPERATURE INCREASE. Zebroski, E.L. (to Electric Power Research Inst., Palo Alto, CA (USA)); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,822,918/A/. 7 Dec 1978. 18p. (In German).

The performance of a fast breeder cooled with Na can be reduced independent of the operating and control mechanism. A thermal expansion body is used that is susceptible to temperature increase. It is arranged between the controlrod drive bar and the controlrod and, for example, made of B₄C. The thermal expansion body is in principle a part of the drive bar and consists of some coaxial cylinders of different radiuses. Their main axes are arranged parallel to the moving direction of the drive bar. The primary cooling medium can flow through them. The single cylinders have alternately high and low thermal expansion coefficients, they consist of stainless steel or of Invar. The thermal expansion body, together with the absorber, has a negative reactivity temperature coefficient and stabilizes the reactor during its operation.

- 3364 IMPROVEMENTS IN OR RELATING TO HEAT EXCHANGERS. Taylor, P.A. (to Babcock and Wilcox Ltd., London (UK)). British Patent 1,534,681/A/. 6 Dec 1978. 4p.

According to the present invention there is provided a method of producing superheated steam by use of the heat in liquid sodium, in which liquid sodium is caused to flow through a space having boundaries of which no part is common with the boundaries of a space in which vapour is produced, a fluid that is inert to sodium is heated by heat exchange at the boundaries of the space through which the liquid sodium flows and serves as the heating medium for the production of vapour, and the vapour is subsequently heated to the final degree of superheat by heat exchange with liquid sodium in a space that has a common boundary with a space through which liquid sodium is passed.

- 3365 COPROCESSING: AN EVALUATION OF CHEMICAL REPROCESSING WITHOUT PLUTONIUM SEPARATION. Pobereskin, M.; Kok, K.D.; Madia, W.J. (Battelle Memorial Inst., Columbus, OH). Nucl. Technol.; 41: No. 2, 149-167 (Dec 1978).

The technical feasibility of a coprocessing concept involving recovery of all the actinides

in the spent fuel as a product group has been analyzed. It has been shown that this can be accomplished by a simple modification of the Purex process. The recovered actinide product group can be reconstituted as a fuel for recycle in either light water reactors (LWRs) or liquid-metal fast breeder reactors (LMFBRs), either by addition of moderately enriched uranium for the LWR case or by controlled partial partitioning of uranium in the LMFB case. Partial partitioning of uranium from a uranium-plutonium extract (that may contain other transuranics, especially neptunium) can be carried out under Purex process conditions that preclude separation of plutonium. A steady-state fuel composition is approached in eight cycles (40 yr) for the LWRs and five cycles (20 yr) for the LMFBs. Potential for proliferation can be greatly reduced for subnational diversion since the plutonium is not separated from its actinide homologs, nor is the recovered actinide fuel fully decontaminated from fission products. The possibility of proliferation by national diversion can be impeded. Recycle of the actinides reduces, via transmutation, the cumulative amount of actinides produced, defers the bulk of the actinide waste disposal to the end of the useful fuel lifetime, and ameliorates the high-level waste management problem.

- 3366 LOW-DECONTAMINATION APPROACH TO A PROLIFERATION-RESISTANT FUEL CYCLE. Asquith, J.G.; Grantham, L.F. (Rockwell International, Canoga Park, CA). Nucl. Technol.; 41: No. 2, 137-148 (Dec 1978).

To prevent the diversion of nuclear material from power production to weapon production either by a nation or by clandestine groups within a nation, the nuclear fuel cycle must be proliferation-resistant and safeguarded. Potentially proliferation-resistant and safeguarded fuel cycles based on low-decontamination pyroprocessing have been developed for the light water reactor (LWR), fast breeder reactor (FBR), and FBR-LWR combination. The major penalty for recycling fission products to the LWR is that fuel enrichment must be somewhat greater to overcome parasitic fission product absorption of neutrons. In the FBR, the major penalty is a slight reduction in breeding ratio due to the displacement of fertile material by fission products. Preliminary cost analysis indicates that these fuel cycles are economically competitive with fuel cycles using conventional reprocessing or those using virgin uranium if spent fuel storage costs are considered.

- 3367 SOME PLUTONIUM RECYCLE PROGRAM FUEL CYCLES HAVING REDUCED PROLIFERATION CHARACTERISTICS. Eschbach, E.A. (Battelle-Pacific Northwest Labs., Richland, WA). Nucl. Technol.; 41: No. 2, 168-179 (Dec 1978).

A review of the plutonium fuel recycle program (1955-1967) did not reveal an economically compelling alternative fuel cycle to slightly enriched uranium, plutonium recycle, or the plutonium breeder. The review included systems involving no chemical separations compared with slightly enriched uranium once through. Nor did freestanding thorium systems appear economic, although synergisms between uranium and thorium may be worth considering, including reduced-density thorium fuels with high fuel durability as an alternative to uranium load following and peaking fuel. The crossed progeny system involving ²³³U enrichment of uranium in light water reactors (LWRs) and conversion of the plutonium to ²³³U in fast reactors may offer a method of providing a high-performance

denatured LWR fuel for the period beyond the availability of economic slightly enriched uranium.

- 3368 SWELLING IN MIXED-OXIDE FUEL PINS. Zimmermann, H. (Kernforschungszentrum, Karlsruhe, Germany). Nucl. Technol.; 41: No. 3, 408-410 (Dec 1978).

The swelling of mixed oxide fuel elements irradiated within the irradiation program of the German Fast Breeder Project was determined. In fuel pins with a burnup-independent high power level, a constant swelling rate of 0.5% per percent burnup was found at burnups above 4%. It is concluded that this swelling rate is induced by solid fission products. The volume-averaged swelling due to fission gases seems to be saturated at a value of approx. 6%.

- 3369 URANIUM REQUIREMENTS FOR ADVANCED FUEL CYCLES IN EXPANDING NUCLEAR POWER SYSTEMS. Banerjee, S. (McMaster Univ., Hamilton, Ont.); Tamm, H. Nucl. Technol.; 41: No. 2, 244-251 (Dec 1978).

When considering advanced fuel cycle strategies in rapidly expanding nuclear power systems, equilibrium analyses do not apply. A computer simulation that accounts for system delay times and fissile inventories has been used to study the effects of different fuel cycles and different power growth rates on uranium consumption. The results show that for a given expansion rate of installed capacity, the main factors that affect resource requirements are the fissile inventory needed to introduce the advanced fuel cycle and the conversion (or breeding) ratio. In rapidly expanding systems, the effect of fissile inventory dominates, whereas in slowly expanding systems, conversion or breeding ratio dominates. Heavy-water-moderated and -cooled reactors, with their high conversion ratios, appear to be adaptable vehicles for accommodating fuel cycles covering a wide range of initial fissile inventories. They are therefore particularly suitable for conserving uranium over a wide range of nuclear power system expansion rates.

- 3370 ANALYZING CONTAINMENT LEAKAGE FROM A SODIUM FIRE BY THE RESPONSE SURFACE METHOD. Person, L.W. (Argonne National Lab., IL). Nucl. Technol.; 41: No. 2, 252-256 (Dec 1978).

The SPOOL-FIRE code has been used with the response surface method and a Monte Carlo simulation to study sodium fire accidents. The study provides a simple method of estimating the radioactive release via containment leakage; the sensitivity of the output consequences to the variations in the input parameters is also presented.

- 3371 ASSESSMENT OF THE CONSEQUENCES OF BOILING IN THE RADIAL BLANKET DUE TO HYPOTHETICAL MAJOR PIPE LEAKS. Kadambi, N.P.; Tilbrook, R.W. (Westinghouse Electric Corp., Madison, PA); Nucl. Technol.; 41: No. 3, 276-282 (Dec 1978).

Boiling initiation in a liquid-metal fast breeder reactor (LMFBR) has in the past been assumed to lead inevitably to the potential for loss of coolable geometry. To ensure conservatism, it was necessary to preclude boiling under all accident conditions. Limited boiling in the radial blanket of the Clinch River Breeder Reactor due to a hypothetical major leak in the primary heat transport system is not likely to lead to assembly-wide dryout and cladding melting. A series of scoping calculations based on applicable physical

processes has shown that (a) boiling is likely to be limited to only six subchannels, (b) flow reversal is unlikely, (c) there are ample heat sinks for condensation of sodium vapor, (d) film dryout is unlikely, and (e) cladding melting is unlikely. The consequences listed are of continuously decreasing likelihood, hence providing confidence that coolable geometry is not threatened by limited boiling in the radial blanket. This analysis was performed for a conventional LMFBR core arrangement.

- 3372 SAS3D ANALYSIS OF UNPROTECTED LOSS-OF-FLOW TRANSIENTS FOR 1200-MW(ELECTRIC) LIQUID-METAL FAST BREEDER REACTOR HOMOGENEOUS AND HETEROGENEOUS CORE DESIGNS. Shih, T.A.; Temme, M.I. (General Electric Co., Sunnyvale, CA). Nucl. Technol.; 41: No. 3, 312-322 (Dec 1978).

A safety comparison was made for two 1200-MW(electric) liquid-metal fast breeder reactor cores with homogeneous and heterogeneous fuel arrangements, respectively. The two cores were conceptually designed to be identical except for those parameters affected by different fuel arrangements. The comparison was limited to the issue of initiating phase energetics in the hypothetical core disruptive accident. Both cores were assumed to be at end-of-equilibrium cycle and subject to unprotected loss-of-flow transients. The SAS3D code was used for analyses with four sets of phenomenological assumptions at different degrees of conservatism. Results of the four corresponding cases showed that the heterogeneous core consistently behaved more mildly than the homogeneous core due to its relatively much lower effective sodium voiding reactivity worth.

- 3373 RESPONSE OF OXIDE FUEL TO SIMULATED THERMAL TRANSIENTS. Bandyopadhyay, G. (Argonne National Lab., IL). Nucl. Technol.; 41: No. 3, 349-358 (Dec 1978).

Direct electrical heating (DEH) experiments have been analyzed to study the fuel and fission gas response to simulated thermal transient conditions in unirradiated UO_2 and irradiated mixed-oxide fuels. The results indicate that both fission gas and fuel melting strongly influence the fuel response behavior during transients. The information obtained from the DEH tests has been used to draw a preliminary fuel response diagram that contributes to an understanding of such macroscopic fuel behavior as gross fuel swelling and fuel failure as functions of the transient time and heating rate. A comparison of these test results and the trend depicted by the fuel response diagram with results obtained from experiments performed in the transient reactor test treat facility indicates that certain aspects of fuel behavior during transients can be reasonably simulated by the DEH technique.

- 3374 NICKEL AND NITROGEN ION IRRADIATION INDUCED VOID SWELLING AND DEFECT MICROSTRUCTURES IN Ni--Al, Ni--Cr AND Ni--Ti SOLID SOLUTIONS. Pinizzotto, R.F. Jr. (IBM Corp., Yorktown Heights, NY); Chen, L.J.; Ardell, A.J. Metall. Trans., A; 9: No. 12, 1715-1727 (Dec 1978).

The effects of solute content, temperature, and irradiation dose on the void swelling characteristics of pure nickel and several nickel base solid solutions (Ni--Al and Ni--Ti containing up to 8 at. pct solute and Ni--Cr containing up to 16 pct Cr) have been investigated. Samples were irradiated in the temperature range 400 to 650°C to a maximum

dose of 70 dpa using 3.5 MeV $^{58}\text{Ni}^+$, 400 keV $^{14}\text{N}^+$ and 400 keV $^{14}\text{N}_2^+$ ions. The irradiation induced microstructures were studied using transmission electron microscopy. The addition of Al, Cr or Ti to Ni is found to decrease the void swelling and mean void diameter and to increase the dislocation density. The behavior of the void number density, $N/\text{sub } v/$, as a function of solute content is found to be dependent upon the irradiation conditions as well as the particular solute addition. $N/\text{sub } v/$ passes through a maximum at approximately 2 pct solute content for Ni--Al and Ni--Cr alloys irradiated at 550°C, but through a minimum at 4 pct for Ni--Ti alloys irradiated at 550 and 600°C. $N/\text{sub } v/$ decreases monotonically as a function of Al content at 600 and 650°C. The results are discussed in terms of recent theories of void swelling suppression due to impurity or solute additions and in light of several correlations between void swelling and material parameters. The behavior of $N/\text{sub } v/$ is found to be best described by the actions of two competing processes. The first enhances void nucleation, is not strongly temperature dependent and is dominant at low solute contents. The second suppresses void swelling, is probably diffusion controlled and dominates in the more concentrated alloys.

- 3375 SENSITIVITY THEORY FOR REACTOR THERMAL-HYDRAULICS PROBLEMS. Oblow, E.M. (Oak Ridge National Lab., TN). Nucl. Sci. Eng.; 68: No. 3, 322-337(Dec 1978).

A sensitivity theory based on reactor physics experience was successfully developed for a reactor thermal-hydraulics problem. The new theory is derived for the case of nonlinear transient heat and mass transfer in a typical reactor subassembly. Suitable adjoint equations for heat and fluid flow are presented along with methods for deriving the sources and boundary and final conditions for these equations. Expressions for the sensitivity of any integral temperature response to problem input data are also presented. The theory is applied to a sample problem describing the steady-state thermal-hydraulic conditions in a Clinch River Breeder Reactor fuel channel. For this case, sensitivity coefficients are derived for several thermal response functions (i.e., peak clad and peak fuel temperature) for all physical input data (i.e., the heat transfer coefficient, thermal conductivities, etc.). A typical uncertainty analysis for peak clad and peak fuel temperature was also performed using uncertainty information about the physical data. Conclusions are drawn about the applicability of this approach to more general problems, and the procedures for its implementation in conjunction with large safety or thermal-hydraulics codes are outlined. The method is also compared with currently used response surface techniques.

- 3376 ROLE OF RAYLEIGH--TAYLOR INSTABILITIES IN FUEL-COOLANT INTERACTIONS. Cooper, F.; Dienes, J. (Los Alamos Scientific Lab., NM). Nucl. Sci. Eng.; 68: No. 3, 308-321(Dec 1978).

The growth of Rayleigh--Taylor instabilities following the deceleration of fuel by a less dense coolant is investigated using the method of generalized coordinates, which permits the study of the nonlinear, late-time aspects of the problem as well as the possibility of fuel freezing at the interface. The authors consider liquid coolant in contact with three possible states of fuel--pure liquid, pure solid, and liquid fuel freezing at the interface--and treat several acceleration mechanisms. For the case of UO_2 and sodium, they follow the growth of the fastest growing wavelength instability

for different acceleration mechanisms and determine the impulse needed for instabilities to grow when freezing is occurring at the interface.

- 3377 COMPARATIVE ANALYSIS OF THREE OF ERDA'S MAJOR R AND D PROGRAMS. Weyant, J.P. (Harvard Univ., Cambridge, MA (USA)). Energy (Oxford); 3: No. 6, 701-735(Dec 1978).

The present model simulates the dynamics of the evolution of the total energy system by requiring both existing and new technologies to compete for introduction on a cost-competitive basis that considers the time phasing of (1) retirement of energy conversion facilities, (2) growth in end-use demands, and (3) escalation of the costs of extracting depleting domestic energy resources. This approach contrasts with a static model used by the Energy Research and Development Administration, wherein assumptions must be made for each future year of interest for (1) maximum capacity constraints for alternative types of conversion facilities and (2) the cost of energy resources. The present model is used to compare the relative consequences and merits of the technology products from the following three of ERDA's major energy R and D programs: (1) the liquid-metal fast breeder reactor (LMFBR), (2) synthetic fuels derived from coal and oil shale, and (3) improved efficiencies for end-use devices (e.g. space heaters).

- 3378 COMPARATIVE ANALYSIS OF THREE OF ERDA'S MAJOR R AND D PROGRAMS. Weyant, J.P. (Harvard Univ, Cambridge, Mass). Energy (Oxford); 3: No. 6, 701-735(Dec 1978).

The model used simulates the dynamics of the evolution of the total energy system by requiring both existing and new technologies to compete for introduction (i.e. commercialization) on a cost-competitive basis that considers the time of phasing of (1) retirement of energy-conversion facilities, (2) growth in end-use demands, and (3) escalation of the costs of extracting depleting domestic energy resources. The model is used to compare the relative consequences and merits of the technology products from the following three of ERDA's major energy R and D programs: (1) the liquid-metal fast breeder reactor (LMFBR), (2) synthetic fuels derived from coal and oil shale, and (3) improved efficiencies for end-use devices (e.g. space heaters). It is found that the development of synthetic fuels derived from coal and oil shale is the only alternative (of the three considered) that provides energy independence for the United States in the next fifty years.

- 3379 FISSION AND FUSION REACTORS. CRITICAL ISSUES FACING THE LONG TERM DEPLOYMENT OF FISSION AND FUSION BREEDER REACTORS. Kulcinski, G.L. (Wisconsin Univ., Madison (USA). Dept. of Nuclear Engineering). Atomwirtsch., Atomtech.; 13: No. 12, 582-589(Dec 1978).

From Reactor congress; Hannover, Germany, F.R. (4 - 7 Apr 1978).

The paper discusses whether breeder reactors should be based on the fission or the fusion concept, taking into account aspects of safety, acceptability to environment and society, dependence on fuel, installation costs, complexity of the technology applied, and earliest possible availability.

- 3380 INTERNAL FLUID FLOW MANAGEMENT ANALYSIS FOR CLINCH RIVER BREEDER REACTOR PLANT SODIUM PUMPS. Cho, S.M.; Zury, H.L.; Cook, M.E.;

Fair, C.E. (Foster Wheeler Energy Corp, Livingston, NJ). Am. Soc. Mech. Eng., [Pap.]; No. 78, 1-12(Dec 1978).

The Clinch River Breeder Reactor Plant (CRBRP) sodium pumps are currently being designed and the prototype unit is being fabricated. In the design of these large-scale pumps for elevated temperature Liquid Metal Fast Breeder Reactor (LMFBR) service, one major design consideration is the response of the critical parts to serve thermal transients. Therefore, the design/development program includes tasks to assure adequate performance as well as reliable operation. In order to demonstrate sufficient margin for operability (i.e. against bearing seizure), a detailed thermal/structural analysis of the critical parts of the pump is required. Analytical and water test data were compared. This paper presents the details of the pump internal hydraulic analysis, and a comparison between analysis and half-scale model test results.

- 3381 MAINTENANCE ASSESSMENT FOR ADVANCED NUCLEAR POWER PLANTS. Basham, S.J. Jr.; Waddell, J.D.; Robinson, R.A.; Kok, K.D. (Battelle Mem Inst, Columbus Lab, Ohio). Am. Soc. Mech. Eng., [Pap.]; No. 78-WA/NE-1, 1-7(Dec 1978).

Proper planning for maintenance and inspection tends to increase the overall availability of nuclear power plants. High potential availability is the basic measure of a successful power plant since replacement power costs are high. The study reported here compared the maintenance and inspection requirements of generic pool and loop type Liquid Metal Fast Breeder Reactors to determine if there were significant differences in these two systems owing to their fundamental difference in design. The size, number and types of failed components affected downtime for maintenance but accessibility and arrangement of components also strongly influenced the downtime. 10 refs.

- 3382 VALUE OF PROTOTYPE TESTING IN THE DEVELOPMENT OF THE IN-VESSEL HANDLING MACHINE FOR FFTF. Sharbaugh, J.E. (Westinghouse Electr Corp, Madison, Pa). Am. Soc. Mech. Eng., [Pap.]; No. 78-WA/NE-3, 1-4(Dec 1978).
Refueling system components for Liquid Metal Fast Breeder Reactors require a high degree of accuracy because of the need to function remotely without visual access. Also, they must have proven reliability to assure minimum plant outage for reactor refueling. The experience gained in the development of the in-vessel handling machine (IVHM) for the Fast Flux Test Facility reactor dramatically indicates that construction, testing, and modification of the design are an essential part of the development of this type of equipment. It is suggested that design and development schedules for this type of equipment must allow for production and testing of prototypes to assure that functional requirements of plant equipment will be achieved.

- 3383 CLOSED LOOP IN-REACTOR ASSEMBLY (CLIRA)-- A FAST FLUX TEST FACILITY TEST VEHICLE. Oakley, D.J. (Westinghouse Hanford Co, Richland, Wash). Am. Soc. Mech. Eng., [Pap.]; No. 78-WA/NE-6, 1-15(Dec 1978).
The Closed Loop In-Reactor Assembly (CLIRA) is a test vehicle for in-core material and fuel experiments in the Fast Flux Test Facility (FFTF). The CLIRA is a removable/replaceable part of the Closed Loop System (CLS) which is a sodium coolant system. The primary purpose of the CLIRA is to provide a test vehicle which

will permit testing of nuclear fuels and materials at conditions more severe than exist in the FTR core and to isolate these materials for the reactor core. This paper characterizes the elements that comprise the CLIRA, and presents the requirements that established the design, describes the design, and evaluates the design conformance to the requirements. 7 refs.

- 3384 INELASTIC STRUCTURAL ANALYSIS: DESIGN IMPLICATIONS AND EXPERIENCE. Griffin, D.S. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). Nucl. Eng. Des.; 51: No. 1, 11-21(Dec 1978).
The current status of inelastic structural analysis is reviewed relative to the needs and requirements for design of elevated temperature nuclear reactor components. The classes of inelastic problems that can be solved in a practical way are identified and the overall scope of ongoing validation programs is reviewed. Three classes of problems are identified where the further development of simplified analysis techniques is necessary to make them tractable for design application.
3385 ESTABLISHING CONSTITUTIVE EQUATIONS FOR USE IN DESIGN OF HIGH-TEMPERATURE FAST-REACTOR STRUCTURES. Pugh, C.E. (Oak Ridge National Lab., TN (USA)). Nucl. Eng. Des.; 51: No. 1, 23-27(Dec 1978).
The presentation describes the approach being used to establish constitutive equations for wide use in the design of fast breeder reactor (FBR) components in the US. The approach combines exploratory experiments, constitutive model studies, studies of computational techniques, and tests of simple structural configurations. Short-time (elastic-plastic) behavior, long-time (creep) behavior, and their interactions are considered, and some of the background to equations now identified for use in current FBR design applications involving three structural alloys is discussed. Comments are also given on current efforts aimed at identifying improved constitutive equations for these alloys and on properties data required for design applications. References are cited which have addressed the status of the process at various times.

- 3386 SNR-300, STRATEGY AND STATE OF THE ELEVATED TEMPERATURE DESIGN. Bieniussa, K. (Interatom, 5060 Bergisch Gladbach 1, Germany, F.R.). Nucl. Eng. Des.; 51: No. 1, 81-84(Dec 1978).
This paper describes the procedure for the elevated-temperature structural analysis of the SNR-300. The preliminary analysis based on elastic calculations will be followed by detailed inelastic analysis, if required to provide assurance against all failure modes. In coherence with the progress of the project, experimental work is being done as a first phase of preparation to the elevated temperature design. On the one hand material data will be established and on the other hand analytical work for computer codes and design criteria will be done. The first inelastic investigations are running. Changes in the geometry of the components and/or the planned operating conditions can be considered in case the structural adequacy cannot be demonstrated even through detailed inelastic analysis.

- 3387 NO FAST BREEDER FOR THE FEDERAL REPUBLIC OF GERMANY. THE PROBLEM OF A TENDENCY TOWARDS NON-LICENSING. Fischerhof, H. (Frankfurt Univ. (Germany, F.R.)). Atomwirtsch., Atomtech.; 13: No. 12, 566-569(Dec 1978).

(In German).

The proposal by North Rhine-Westphalian State Minister of Economics Riemer to modify the SNR-300 prototype fast breeder power plant under construction at Kalkar since 1973 in such a way that the use of thorium instead of uranium-238 would no longer breed plutonium but consume it, has aroused considerable attention, chiefly also for the reason that the Minister no longer wants to issue the pending partial construction permit and the subsequent operating permit for the present fast breeder concept. In addition to technical and scientific implications this attitude has also raised legal questions. They above all relate to the question to what extent the basic decision of 1972, when the first construction permit was issued, is binding upon the licensing authority, both in terms of energy law and atomic law, and to what extent the Federal Government, which supports and funds most of the development of fast breeder reactors, has a right to issue directives in order to have the pending permit granted.

3388 CONSTRUCTION LICENSE FOR MECHANICAL ENGINEERING AND ELECTRICAL ENGINEERING COMPONENTS OF THE SNR. Bundesanzeiger; 30: No. 243, 2(Dec 1978). (In German).

CONSTRUCTION/ELECTRICAL EQUIPMENT; REACTOR LICENSING; SNR-1 REACTOR

3389 PROPOSED MECHANISM FOR INTERNAL CLADDING CORROSION IN LMFBR MIXED OXIDE FUEL PINS. Bradbury, M.H.; Pickering, S.; Whitlow, W.H. (Commission of the European Communities, Karlsruhe (Germany, F.R.). European Inst. for Transuranium Elements). J. Nucl. Mater.; 78: No. 2, 272-280(Dec 1978).

A model is proposed which describes a possible chronological sequence of events during inner cladding corrosion in mixed oxide fuel pins. The corrosion reaction is visualized as being primarily chemical in character, involving the cladding steel, the fuel and the more aggressive fission products, notably caesium in the presence of oxygen. The model attempts to explain how corrosion starts, how it depends on the oxygen potential, why it occurs non-uniformly; also covered are phase changes within the cladding steel and morphological features such as the intergranular form of attack and the distribution of corrosion products in the fuel/cladding gap.

3390 DEVELOPMENT OF A BEARING FOR SODIUM PUMPS. Gelijon, H.C. Energiespectrum; 2: No. 12, 296-301(Dec 1978). (In Dutch).

For the development of the prototype of a circulation pump for the fast breeder reactor SNR 300, a method is required to run a support bearing for a vertical rotor, about 7m long, working in a liquid sodium environment. It is convenient in such a situation to select the process medium as the lubricant. Liquid sodium from 500° to 600° has however, no attractive properties as a lubricant. The dynamic viscosity is a factor 4 to 5 lower than water at room temperature, so that for a hydrodynamic pressure build-up in the lubricant film, there is little prospect. A hydrostatic layer may be present. Such a bearing derives its carrying capacity from the supply of lubricant under pressure. For the case of the circulator pump, the feed to the bearing can be made from a tapping in the pump diffuser. As a contrast for the Nuclear Energy Project Group TNO and the BV Nerotoom, the Institute TNO for Machine Tool Constructions (IWeCo) has developed a new type of hydrostatic bearing.

3391 CONDITIONING OF LIQUID SODIUM PUMPS BY THE USE OF ACOUSTIC METHODS. Moelker, W.H.; ten Wolde, T. Energiespectrum; 2: No. 12, 302-303(Dec 1978). (In Dutch).

In pumps, cavitation is often a serious source of erosion, and can lead to negative effects such as a shortened lifetime for the rotor, a reduced performance of the pump and a deterioration in the reliability. For liquid sodium pumps such as those as for example, that have been made for the nuclear power station at Kalkar, cavitation must be seen as a phenomenon which can be reduced and in every case kept under scrutiny. It is consequently necessary to detect cavitation in liquid sodium pumps. This is no simple task as liquid sodium is non-transparent, the temperature is high (up to 650°C) the rotor blade cavities are rotating and cavitation is an undeveloped physical phenomenon. Nevertheless, there is a possibility of using relatively simple aids such as accelerometers and mechanical sensors to achieve a fairly reliable insight into the cavitation features within the pump. A contract from Neratoom has been undertaken by the Technical Physics Service of the TNO (TPD) using various detection methods.

3392 TRANSIENT FLOW AND HEAT TRANSFER OF LIQUID SODIUM COOLANT IN THE OUTLET PLENUM OF A FAST NUCLEAR REACTOR. Markatos, N.C.G. (CHAM Ltd., Wimbledon (UK)). Int. J. Heat Mass Transfer; 21: No. 12, 1565-1579(Dec 1978).

A calculation procedure for axisymmetric elliptic flows is applied to predict the transient velocity and temperature fields of a heavy fluid jet issuing vertically into a volume of relatively light fluid. This situation arises in the outlet plenum of a Liquid-Metal-Cooled Fast Breeder Reactor (LMFBR) during reactor transients. The time averaged conservation equations for momenta and heat transfer were solved on a CDC6600 digital computer for various plenum inlet transients, along with a two-equation model of turbulence and proper modelling of the buoyancy terms. Predictions are presented of flow and heat transfer in the form of velocity vector plots and temperature contours. Predictions are in qualitative agreement with expectations, invariably establishing that the flow by-passes the outlet plenum.

3393 CONTROL OF TRITIUM IN LIQUID-METAL-COOLED FAST BREEDER REACTORS (LMFBRs). McGuire, J.C. (Hanford Engineering Development Lab., Richland, WA (USA)); Renner, T.A. (Argonne National Lab., IL (USA)). At. Energy Rev.; 16: No. 4, 657-695(Dec 1978).

In liquid-metal-cooled fast breeder reactors tritium originates principally in two locations, the fuel and the control rods. It is created in the fuel by ternary fission and in control rods by neutron interaction with the boron in B₄C. Tritium diffuses readily through the cladding material of these components and is carried by the sodium coolant throughout the primary circuit. From primary sodium it may diffuse through heat exchangers to the secondary circuit, be released to the cover gas, permeate reactor walls into the containment vessel, or be removed by the primary cold trap. Similarly, from the secondary sodium tritium may diffuse through the heat transfer surfaces of the steam generator, be released to the cover gas, permeate through reactor walls to auxiliary containment, or be removed by the secondary cold traps. Protective oxide films on the outside walls of the reactor vessels and sodium piping significantly reduce the permeation rate

of tritium through the structural materials. In addition, tritium retention within the sodium containment is enhanced by coprecipitation with hydrogen in the cold traps. A substantial hydrogen influx to the secondary circuit as a result of steam-side corrosion and permeation through steam generator tubing provides a source of hydrogen that assists in tritium control. Once in the secondary circuit, hydrogen is either removed by the secondary cold traps or diffuses through the intermediate heat exchanger to the primary circuit. The hydrogen source term in a reactor is roughly 10^4 times greater than that for tritium. Therefore, in either the primary or the secondary circuit, sufficient hydrogen is available for efficient tritium removal by coprecipitation.

- 3394 Donen Ju-nenshi. (TEN YEARS' HISTORY OF POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION). Tokyo, Japan; Power Reactor and Nuclear Fuel Development Corp. (Dec 1978). 636p. (In Japanese).

Power Reactor and Nuclear Fuel Development Corporation (PNC) has advanced steadily for ten years since its establishment along the policy lines of Japan's own nuclear energy development. Brilliant results thus include: the world's fifth nation of possessing an experimental fast reactor, completion of a (prototype) heavy-water power reactor, establishment of uranium ores prospecting and refining technology, successful approach to uranium enrichment, and fuel reprocessing and waste management leading to Japan's nuclear fuel cycle. Intended for further efforts by PNC personnel and for report to the those people concerned, the present volume of PNC's ten years' history is in chapters of power reactors development, nuclear fuel development, safety management, general administration, and safeguards and nuclear materials control.

- 3395 LIQUID METAL FAST BREEDER REACTORS. Khodarev, E. (International Atomic Energy Agency, Vienna (Austria). Div. of Nuclear Power and Reactors). Int. At. Energy Agency Bull.; 20: No. 6, 29-36 (Dec 1978).
The present state and trends in the development of LMFBR type reactors are analyzed. General principles of fast breeder reactors, main fast breeder reactor types and their design parameters are discussed.

- 3396 POSSIBILITY OF ACOUSTIC DETECTION ON SODIUM BOILING IN A FAST REACTOR BY A PULSE SYSTEM. Kebabze, B.V.; Aleksandrov, K.A.; Golovanov, V.V. At. Energ. (USSR); 45: No. 6, 461-463 (Dec 1978). (In Russian).
ACOUSTIC MONITORING; BOILING; BACKGROUND NOISE; BOILING DETECTION; FAST REACTORS; PULSE TECHNIQUES; SODIUM COOLED REACTORS

- 3397 CREYS-MALVILLE POWER PLANT. Rev. Gen. Nucl.; No. 6, 474-475 (Dec 1978). (In French).
This article presents the power plant construction works, which were started on the Creys-Malville area at the beginning of 1977, after the preparing of the site. Information is given on the different kinds of works going on, among which, for example, the construction of the reactor pressure vessel.

- 3398 (RISLEY-Trans--4105) MONITORING THE CONDITION OF SODIUM PUMPS USING ACOUSTIC METHODS. Moelker, W.H.; ten Wolde, T. Translated from Energiespectrum; 302-303 (Dec 1978). 6p. Dep. NTIS (US Sales Only), PC

A02/MF A01.

Cavitation is a serious source of erosion in pumps which can result in negative effects such as reduced impeller life, reduced pump performance and reduced reliability. For sodium pumps such as those which will form a part of the Kalkar Nuclear Power Station, for instance, cavitation must therefore be regarded as a phenomenon which must be avoided so far as is possible and which in any case must be monitored. Methods for detecting cavitation in sodium pumps are described.

- 3399 REFUELING DEVICE FOR A NUCLEAR REACTOR WITH FAST NEUTRONS. Artemev, L.N.; Vitalevitsch, K.V. (to Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,723,680/A/. 30 Nov 1978. 17p. (In German).

The refueling device has got a drum-shaped cartridge containing cells for unirradiated fuel assemblies and control rods as well as a grab for manipulation. The cells for the fuel assemblies are identical with those for the control rods. For the control rods in stock and those on the way from the drum-shaped magazine to the refueling mechanism there are provided cup-shaped casings (adapters) adapting their ends to the dimensions of the fuel assemblies. The casing is clamped to an annular tee-slot below the top by means of a collet. By this means fuel assemblies as well as control rods can be handled with the grab. By means of a stop on the grab the casing can be detached from the control rod.

- 3400 DEVICE FOR THERMAL PROTECTION OF A NUCLEAR REACTOR VESSEL. Gama, J.M.; Paillard, L. (to Commissariat a l'Energie Atomique). US Patent 4,127,444. 28 Nov 1978. Priority date 6 Apr 1976, France, 10p.

A part of the cold liquid metal which is injected at the base of the reactor core is circulated along the bottom portion of the main reactor vessel and guided by a shell which is concentric with the vessel. A mass of inert gas is trapped within a leak-tight annular chamber above the liquid metal. During reactor operation, the gas pressure is adjusted in order to maintain the gas/liquid metal interface at a substantially constant level and to obtain along the vessel a temperature distribution which ensures limited amplitude of stress variation.

- 3401 CONTROL ROD DRIVE FOR A NUCLEAR REACTOR. Schutachkov, I.I.; Gorunov, V.S.; Zalcev, B.I.; Derevyankin, N.E.; Petrov, V.A.; Istomin, S.D.; Kovalentschik, D.I.; Archipov, E.A.; Serebryakov, V.I.; Katschaln, V.S. (to Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,722,050/A/. 23 Nov 1978. 32p. (In German).

The control of the rod repositioning gear unit and the control unit of the profile grab of the control rod drive for the alkali metal-cooled fast breeder reactor is achieved by an electromotor being arranged outside the hermetic drive casing. The guide tube is directly repositioned by the rod repositioning gear unit. Coupling control of the drive with the control rod is done in the lower operative position of the control rod and that because of the interaction of the tie rod arranged on the spring-mounted control rod with the induction transmitter for the lower position of the control rod. In the transfer position the rod is fixed within the guide tube.

- 3402 REFUELING FACILITIES. Daiku, M. (to

Toshiba Corp., Kawasaki, Kanagawa (Japan)). Japanese Patent 1953-132,690/A/. 18 Nov 1978. 3p. (In Japanese).

Purpose: To provide the subject facilities by which fuel or core assemblies together with coolants are charged and discharged in a state where the fuel and core assemblies are placed in a pot, thereby preventing the reactor coolants from overflowing and falling from and in the reactor. Constitution: The pot in a state where core assemblies and coolants are contained therein is transferred to a pot relaying device outside the reactor by means of a pot charging and discharging device. The pot moves at a predetermined distance by the pot relaying device to a position where only fuel or core assemblies are caught and taken out by the fuel transfer device. On the other hand, the second fuel or core assemblies are carried by means of a fuel transfer device, and placed in the pot, then they being transferred by the refueling machine and inserted into the reactor vessel.

3403 SODIUM-WATER REACTION PRODUCT FLOW SYSTEM. Shirataki, K.; Wada, H. (to Mitsubishi Atomic Power Industries, Inc., Tokyo (Japan)). Japanese Patent 1953-132,695/A/. 18 Nov 1978. 3p. (In Japanese).

Purpose: To provide the subject equipments wherein thermal insulating layers which neither exfoliate nor react by the impact due to high temperature sodium and hydrogen gas and are used for mitigating the thermal impact are provided on the inner surfaces of the emission system equipments, thereby preventing the destruction of the emission system equipments. Constitution: Thermal insulating layers are formed on the inner surfaces of sodium-water reaction product emission system equipments, that is, the inner surface of the emission system pipeline, that of the accommodation vessel and the surface of the cyclone separator, by film treatment, coating or heat resisting coating, and these surfaces are covered with the layers. Each of the layers is made of a material which does not cause a rapid reaction with high temperature sodium or hydrogen gas nor exfoliates and is withstandable for several seconds in which the thermal impact of at least the emission system comes into question, and its thickness is more than one capable of securing the necessary thermal resistance computed by the thermal impact analysis of the emission system.

3404 METHOD OF SAFE OPERATION OF LMFBR TYPE REACTOR. Kondo, R.; Amada, T. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1953-131,394/A/. 16 Nov 1978. 3p. (In Japanese).

Purpose: To provide the subject method of heat exchange between a primary cooling system where a liquid metal circulates and a water-steam system by means of a stable substance which reacts with neither the liquid metal nor water, thereby elevating the workability, reliability and economy of the reactor. Method: The heat transmission tube of a heat exchanger is of a double structure, and the interior of the inner tube of the heat transmission tube forms a flow passage of the water-steam system, and the outer periphery of the outer tube of the heat transmission tube forms a flow passage of the liquid metal which is a primary coolant. The heat exchange between the inner tube of the heat transmission tube is carried out by means of a stable substance such as mercury which reacts with neither water nor the liquid metal which is filled in the annulus section between the inner tube of the heat transmission tube and the outer tube thereof. The stable

substance is circulated in a loop shape as necessary.

3405 NUCLEAR FUEL ELEMENTS. Ozaki, S.; Ochi, Y.; Matsumura, K. (to Toshiba Corp., Kawasaki, Kanagawa (Japan)). Japanese Patent 53-131,397/A/. 16 Nov 1978. 5p. (In Japanese).

The purpose is to use a low swelling alloy, in which the formation of voids in the metal of alloy is suppressed by the action of fast neutrons at high temperatures, as a cladding material in a fuel rod used in a fbr type reactor and a structural material for supporting the fuel core. The fuel core composed of an oxide, a nitride or a carbide of uranium is clad with an austenite stainless steel containing 6 to 26% of chromium, 0.03 to 0.5% of titanium, and less than 0.7% of silicon.

3406 TEMPERATURE-DEPENDENT LIQUID METAL FLOWRATE CONTROL DEVICE. Carlson, R.D. (to Dept. of Energy). US Patent 4,125,221. 14 Nov 1978. Filed date 7 Sep 1977. 4p. PAT-APPL-831,193.

A temperature-dependent liquid metal flowrate control device includes a magnet and a ferromagnetic member defining therebetween a flow path for liquid metal, the ferromagnetic member being formed of a material having a curie temperature at which a change in the flow rate of the liquid metal is desired. According to the preferred embodiment the magnet is a cylindrical rod magnet axially disposed within a cylindrical member formed of a curie material and having iron pole pieces at the ends. A cylindrical iron shunt and a thin wall stainless steel barrier are disposed in the annulus between magnet and curie material. Below the curie temperature flow between steel barrier and curie material is impeded and above the curie temperature flow impedance is reduced.

3407 METHOD FOR PRE-HEATING LMFBR TYPE REACTORS. Yokozawa, A.; Kataoka, H. (to Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Japanese Patent 53-129,798/A/. 13 Nov 1978. 3p. (In Japanese).

Purpose: To enable pre-heating for the inside of the reactor container and the inside of the coolant recycling system with no additional facilities. Method: The coolant recycling system is composed of a heat exchanger, a mechanical pump, a check valve, a flow meter or the like and it is connected in series by way of a pipe line to a reactor container. The mechanical pump is used as a gas recycling device upon pre-heating and it is designed so that a blower such as a fan can be replaced for the impeller of the pump. The inside of the reactor container and the inside of the coolant recycling system is at first filled with an inert gas such as for use with cover gas. Then, nuclear fuels are loaded to attain criticality. Simultaneously, the blower is started and the control rods are operated while cooling the nuclear fuel with the inert gas thus to obtain heat required for pre-heating the pipe line or the like from the nuclear fuels. After the completion of the pre-heating, the liquid metal is charged.

3408 BETHE DECRIES U.S. LAG IN BREEDERS AND NAVY'S NEAR-BREEDER SECRETS. Energy Res. Rep.; 4: No. 20, 2-6(6 Nov 1978).

Part two of an interview with Hans Bethe continues with his projection of uranium

supplies lasting through 2030 and his concern that U.S. policy is retarding development of breeder reactors to the point where imported breeders may be needed. Bethe prefers the liquid-metal breeder reactor concept because of its efficient cooling properties and its low pressure compared to light-water reactors. He feels that safety concerns about the critical mass potential are not warranted, but that the heat-exchanger design is more critical. The discussion covers a comparison of the French program, which he finds to be the most advanced; the Shippingport "near breeder"; and the Canadian program, which he considers the best "near-breeder" if it is modified to use thorium.

- 3409 FAST NUCLEAR REACTOR. Iljunin, V.G.; Kusnezov, I.A.; Murogov, V.M.; Silaev, J.V.; Schmelev, A.N. (to Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,323,938/B/. 2 Nov 1978. 5p. (In German).

The FBR BN-350 has got a reactor core with a set of fuel cartridges. In vertical direction the inner space of the cartridges is divided into two sections containing fuel elements. In order to improve the conversion ratio the lower fuel elements-regarded in flow direction of the coolant - are filled with metallic uranium and the upper fuel elements with ceramic plutonium. The height of the lower section amounts to 0.3 up to 0.7 times the length of a cartridge.

- 3410 USE OF ZERO POWER PLUTONIUM REACTOR MEASUREMENTS AS A SUPPORT OF CRITICALITY PREDICTION FOR THE SNR-300. Pilate, S. (Belgonucleaire Rue du Champ de Mars, Brussels, Belgium); de Wouters, R.; Wehmann, U.; Helm, F.; Scholtyssek, W. Nucl. Technol.; 41: No. 1, 5-11 (Nov 1978).

Evaluations of criticality measurements performed in various SNEAK and Zero Power Plutonium Reactor (ZPPR) cores are compared. The best available methods of calculations (including transport theory) are used. The ZPPR results support well the trend indicated by the SNEAK evaluations for clean cores and for cores with followers; for cores with absorbers partially inserted, the agreement is only rough. Evaluations of control rod worth measurements are therefore also compared, using the routine method of calculation for SNR-300 (diffusion theory). The control rod worths are largely underestimated in SNEAK ($C/E = 0.89$), but only slightly underestimated in the ZPPR ($C/E = 0.97$). The difference in the nature of core fuel (uranium in SNEAK, plutonium in the ZPPR) could be at the origin of this discrepancy.

- 3411 ACTINIDE REDISTRIBUTION IN THE OUTER REGIONS OF MIXED-OXIDE FUEL. Lawrence, L.A.; Hata, D.C.; Washburn, D.F. (Hanford Engineering Development Lab., Richland, WA). Nucl. Technol.; 41: No. 1, 60-70 (Nov 1978).

Significant actinide redistribution was observed in the outer low-temperature region of uranium-plutonium mixed-oxide fuel. Data from the large number of fuel pins examined indicated boundaries within which redistribution in the outer low-temperature regions of the fuel occurred. Plutonium redistribution was not observed in fuel pins with an initial fuel oxygen-to-metal ratio (O/M) of >1.98 or in fuel irradiated to burnups of <5.0 at. %. Fuel pins with an initial O/M ratio of 1.96 exhibited plutonium enrichments in the equiaxed grain region and uranium enrichments in the equiaxed grain region and uranium enrichments on the fuel outer periphery at a burnup of greater than or equal to 5.0 at. %.

Approx. 6.5 at. % burnup, a transition in character of the actinide distribution occurred, resulting in plutonium enrichments on the outer periphery of the fuel. Increasing the fuel initial O/M to 1.97 decreased the burnup at which plutonium enrichment occurred near the equiaxed grain region from 6.5 to 5.0 at. %. Conversely, decreasing the initial O/M ratio from 1.96 to 1.95 increased the burnup at which plutonium enrichment occurred in the equiaxed grain region from approx. 6.5 to approx. = 7.5 at. %.

- 3412 RESISTANCE WELDING OF PRESSURIZED CAPSULES FOR IN-PILE CREEP EXPERIMENTS. de Burbure, S. (Centre d'Etude de l'Energie Nucleaire, Mol, Belgium). Weld. J. (Miami); 57: No. 11, 23-30 (Nov 1978).

Resistance welding was chosen to obtain welds that could qualify in in-pile tests of fuel cans for use in LMFBR reactors to obtain creep behavior information on ferritic steel. Methods and resistance welding equipment are described. (FS)

- 3413 BURNING RATES OF SODIUM POOL FIRES. Newman, R.N.; Payne, J.F.B. (CEGB, Berkeley Nucl Lab, Gloucestershire, Engl.). Combust. Flame; 33: No. 3, 291-297 (Nov 1978).

The burning rates of sodium pool fires were measured at pool diameters of 50, 100, and 220 mm. The pool burning was found to exhibit a flat flame zone residing close to the surface of the liquid metal pool. A model of sodium poolburning is produced dealing in particular with the variation of burning rate with pool size. This analysis forms the basis of prediction of heat release and smoke release rates as may occur from large pool fires following accidents in sodium-cooled fast breeder nuclear reactors.

- 3414 LOAD-VARYING OPERATION OF A NUCLEAR POWER PLANT WITH FAST BREEDER REACTORS. Iljunin, V.G.; Kuznetsov, I.A.; Murogov, V.M.; Shmelev, A.N. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). Kernenergie; 21: No. 11, 341-344 (Nov 1978). (In German).

The possibility of effectively using fast breeders in nuclear power plants with several nuclear reactors of different type is discussed. This possibility may emerge in a nuclear power plant where some nuclear reactors generate heat in a single thermodynamic cycle. In such a nuclear power plant metallic fuel can be used in the fast breeder such increasing the breeding ratio and, at the same time, making possible generation of high temperature heat in the plant.

- 3415 MODEL TO DESCRIBE THE ROLE OF NON-EQUILIBRIUM INTRAGRANULAR FISSION GAS BUBBLES DURING TRANSIENT HEATING. Wood, M.H. (UKAEA, Harwell. Atomic Energy Research Establishment). J. Nucl. Mater.; 78: No. 1, 58-63 (Nov 1978).

The presence of fission gases in nuclear fuel could markedly affect the course of possible fast reactor accidents. Thus, it is important to understand the behaviour of these gases under the various possible accident situations. Recent out-of-pile rapid heating experiments on pre-irradiated fuel have indicated that non-equilibrium intragranular gas bubbles may be formed during a transient. Moreover, it is anticipated that the formation of such overpressured bubbles could produce important consequences for the subsequent transient behaviour. In the present work the author has extended a previous model of the

fission gas behaviour under rapid heating conditions, which considered all gas bubbles to be in equilibrium throughout the transient, for application to possible non-equilibrium regimes. Results of calculations performed with the models show the importance of employing the more extensive theoretical approach.

- 3416 EXPERIMENTAL INVESTIGATION OF THE EFFICIENCY OF INTERMEDIATE HEAT EXCHANGERS OF THREE-LOOP NUCLEAR POWER PLANTS. Karetnikov, G.B.; Rassokhin, N.G.; Mel'nikov, V.N. Teploenergetika (Moscow); No. 11, 62-65 (Nov 1978). (In Russian).

An experimental determination of effective coefficients of overflows and an investigation of mixed convection in the inter-tube space of single-bundle and triple-bundle models of the BN-350 intermediate heat exchangers was carried out. The triple-bundle model consisted of a narrow box-type housing with three vertical U-shaped tube bundles 1 m high. Sodium was used in both loops as the heat carrier. 11 refs.

- 3417 EXTINGUISHER POWDER FOR LIQUID METAL FIRES. Menzenhauer, P.; Ochs, G. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). Brandschutz; 32: No. 11, 318-321 (Nov 1978). (In German).

The extinguisher introduced here based on graphite exhibited considerably improved extinguishing properties compared to other extinguishing powders. It has no aggressive properties, is, as far as could be tested in the short time available, non-hygroscopic and thus very easy to remove after extinguishing and cooling of the fire. The amount of extinguisher necessary is only a fraction of the amounts needed of other common powders. Storage is no problem and merely a small storage amount is required. This extinguisher is excellently suitable for fighting sodium surface fires.

- 3418 EVALUATION OF ANELASTICITY AND PLASTIC CREEP LIMIT FOR ALLOY 800. Tavassoli, A.A. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Service de Recherches Metallurgiques Appliquees). Nucl. Eng. Des.; 50: No. 3, 425-430 (Nov 1978).

The elevated temperature tensile, anelastic and creep properties of a precipitation strengthened Ni-Cr austenitic steel (alloy 800) have been evaluated with respect to LMFBFR application. These properties have been estimated and formulated for small strains and temperatures on the order of 823 K. It has been shown that despite suppression of the in-service stresses to beneath the alloy's yield strength, deformation will occur at such temperatures and the stress below which plastic strain is effectively zero (plastic creep) is negligible for a reactor life time. The plastic creep limit at a given time and temperature markedly depends on the degree of precipitation hardening or prior cold deformation for alloy 800. The contribution from anelastic deformation is less than 0.05% and only becomes important for strict dimensional control and constrained parts.

- 3419 EXPERIMENTS IN SODIUM BOILING ON THE BOR-60 REACTOR. Afanas'ev, V.A.; Aleksandrov, K.A.; Kabadze, B.V. At. Energ. (USSR); 45: No. 5, 338-342 (Nov 1978). (In Russian).

Data on boiling fuel assembly (BFA), measuring system and experimental results on sodium boiling in the BOR-60 reactor core are presented. BFA consists of 18 tungsten rods and

an iron rod of 6 mm in diameter and 450 mm in length. Sodium boiling in the BFA of the BOR-60 core achieved as a result of heating of tungsten rods by γ -radiation. Four regimes of boiling were tested depending on the power rise rate till 10 MW. A conceptual possibility of detecting boiling by acoustic and neutron noises and some results of spectral and correlation analyses are demonstrated. It is shown that signal forming is necessary to increase detection reliability of boiling in the fast reactor core.

- 3420 EXPERIMENTAL DETERMINATION OF VISCOSITY OF SODIUM VAPOR. Sidorov, N.I.; Tarlov, Yu.V.; Yargin, V.S. (Moscow Aviat Inst, USSR). Heat Transfer - Sov. Res.; 10: No. 6, 125-131 (Nov-Dec 1978).

Viscosities of sodium were measured between 1101 to 1186; degree; K and 0.2 to 0.8 bar with an accuracy of $\pm$ 3 percent. The results of their theoretical correlation for the range of 900 to 1300; degree; K and 0.01 to 2 bar are presented. 6 refs.

- 3421 KEY ISSUES OF NUCLEAR POWER R AND D IN JAPAN. An, Shigehiro (Tokyo Univ. (Japan). Faculty of Engineering). Genshiryoku Kogyo; 24: No. 11, 41-46 (Nov 1978). (In Japanese).

Nuclear power technology, spreading over a number of specialized areas, is so-to-speak an extreme technology requiring the cooperation of scientists and technologists in respective fields. Its research and development require tremendous amounts of funds and manpower, with long lead times from the basic research up to the practical use. The nuclear power technology exerts thus large influences both socially and economically. The research and development of nuclear power do not remain only in technology. It is a problem spreading over society, economy, politics and diplomacy and compatibility is required in an extensive national system containing nuclear power. On the research and development of nuclear power in Japan, the following matters are described: targets of nuclear power development and the reality; perfection in LWR technology; FBR development; establishment of the nuclear fuel cycle comprising uranium resources, uranium enrichment, downstream and spent fuel storage; and the funding and program of research and development, cooperation of respective organizations, etc.

- 3422 RESERVATIONS CONCERNING AND PRECISION OF THE ATOMIC ENERGY ACT. THE CONSTITUTIONALITY OF THE FAST BREEDER OF KALKAR. Rengeling, H.W. Neue Jurist. Wochenschr.; 31: No. 44, 2217-2223 (Nov 1978). (In German).

During the constitutional review of section 7 Atomic Energy Act, the principles of reservations and of precision of the law were in the centre of the discussions. The author shows in detail that these standards, due to their constitutional derivation and their concretisation by jurisdiction, did not lead to unconstitutionality of section 7 AtG, as far as this regulation makes the licensing of the nuclear power plant of the type of the so-called fast breeders possible.

- 3423 DEVICE TO CONTROL THE COOLANT FLOW IN FUEL OR FERTILE ELEMENTS OF NUCLEAR REACTORS. Werner, F. (to Siemens A.G., Berlin (Germany, F.R.); Siemens A.G., Muenchen (Germany, F.R.); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,005,868/C/. 26 Oct 1978. 5p. (In German).

Metering the coolant flow through the zone

of fuel or blanket elements is difficult because of the very limited possibility for external impact on the reactor core. In the invention, the coolant feed duct is to be connected with the coolant inlet side of the reactor by means of a variable throttle. The throttle is made up by the foot point of a fuel element; it can be varied by turning this foot point around the axis of the fuel element. Thus, turning of the fuel element in its seat within the support plate allows the flow to be controlled for the group combined in a subassembly and arranged in the same cooling channel. This metering system can be used in all reactor systems, but it offers special advantages when employed in fast breeder assemblies cooled by sodium.

- 3424 NUCLEAR REACTOR REFUELLING SYSTEM. Hopkins, S.R. (to UKAEA). British Patent 1,528,012/A/. 11 Oct 1978. 4p.

A handling system for fuel element assemblies of a liquid metal cooled nuclear reactor is described, the system comprising a transfer bucket and ramp means for guiding the bucket for lifting or lowering in a direction inclined to the vertical. A branch of the ramp means causes a change of attitude of the bucket on the ramp between parallelism and a vertical attitude. Guide means are provided for ensuring that the bucket engages said branch during lifting so as to change its attitude from vertical towards inclined.

- 3425 NUCLEAR REACTORS. Hopkins, S.R. (to UKAEA). British Patent 1,528,272/A/. 11 Oct 1978. 4p.

A nuclear reactor construction is described of the kind comprising a reactor core submerged in a pool of liquid metal contained by a primary vessel. The construction has an inclined chute for guiding a bucket containing a fuel element assembly into and out of the primary vessel and shock absorbing means at the foot of the chute for arresting a falling bucket. There is a vertical shaft displaceable along the longitudinal axis and a horizontally pivoted hooked link arranged so that the link can be pivoted to engage a radial projection on the bucket when in an inclined position and, by upward displacement of the shaft, pivot the bucket about its base into a vertical position.

- 3426 ULTRASONIC TRANSDUCERS. Bishop, J. (to UKAEA). British Patent 1,526,950/A/. 4 Oct 1978. 3p.

An ultrasonic transducer is described for use in a liquid metal environment, comprising piezo-electric material having a metallic coating which is bonded to a nickel diaphragm with lead. The piezo-electric material is lead zirconate titanate, and the metallic coating for the piezo-electric material is of silver.

- 3427 NUCLEAR REACTOR INSULATION AND PREHEAT SYSTEM. Wampole, N.C. (to Dept. of Energy). US Patent 4,118,278. 3 Oct 1978. Filed date 27 Sep 1976. 6p.
PAT-APPL-727,081.

An insulation and preheat system is disclosed for preselected components of a fluid cooled nuclear reactor. A gas tight barrier or compartment of thermal insulation surrounds the selected components and includes devices to heat the internal atmosphere of the compartment. An external surface of the compartment of enclosure is cooled, such as by a circulating fluid. The heating devices provide for preheating of the components, as well as maintenance of a temperature sufficient

to ensure that the reactor coolant fluid will not solidify during shutdown. The external cooling limits the heat transferred to other plant structures, such as supporting concrete and steel. The barrier is spaced far enough from the surrounded components so as to allow access for remote or manual inspection, maintenance, and repair.

- 3428 VARIABLE FLOW CONTROL FOR A NUCLEAR REACTOR CONTROL ROD. Carleton, R.D.; Bhattacharyya, A. (to Dept. of Energy). US Patent 4,118,275. 3 Oct 1978. Filed date 12 Dec 1974. 6p.

PAT-APPL-532,285.

A variable flow control for a control rod assembly of a nuclear reactor that depends on turbulent friction through an annulus. The annulus is formed by a piston attached to the control rod drive shaft and a housing or sleeve fitted to the enclosure housing the control rod. As the nuclear fuel is burned up and the need exists for increased reactivity, the control rods are withdrawn, which increases the length of the annulus and decreases the rate of coolant flow through the control rod assembly.

- 3429 TRANSIENT PRESSURIZATION AND TRANSPORT OF A TWO-COMPONENT GAS WITHIN INTERCONNECTED POROSITY OF SOLID MIXED-OXIDE FUEL. Hofmann, J.R. (Argonne National Lab., IL). Nucl. Sci. Eng.; 68: No. 1, 73-90 (Oct 1978).

A model has been developed to describe the transient pressure field within the interconnected porosity of solid mixed-oxide fast reactor fuel during a reactor transient. The pore gas may be composed of up to two distinct chemical species, so that gas released from fuel grains may differ chemically from the fill gas originally present within the porosity of the fuel. The volume expansion of fuel upon melting is accounted for, but mechanical deformation of the solid fuel is not modeled. Results are presented for a hypothetical unprotected transient overpower accident in a gas-cooled fast reactor with ramp rates of 0.10, 1.0, and 10.0 dollar/s. In these calculations, fuel cladding failure is computed from a linear accumulative damage law and a Larson-Miller parameter correlation.

- 3430 EFFECTS OF NOZZLE-TO-TARGET DISTANCE ON WASTAGE IN SMALL-LEAK SODIUM-WATER REACTIONS. Kanegae, N.; Hashiguchi, K.; Ikemoto, I.; Hori, M. (Power Reactor and Nuclear Fuel Development Corp., Ibaraki, Japan). Nucl. Technol.; 40: No. 3, 261-277 (Oct 1978).

The wastage effects of a sodium-water reaction resulting from a small leak in a liquid-metal fast breeder reactor steam generator were investigated. These experiments were conducted at low steam injection rates, ranging from 0.07 to 13.8 g/s, with the following test conditions: sodium temperature 480°C, sodium velocity 0.24 m/s, and target material 2 1/2% Cr-1 Mo steel. Employed were nozzle diameters (D) from 0.2 to 1.5 mm, nozzle-to-target distances (L) from 5 to 100 mm, and L/D ratios from 7.14 to 400. An empirical equation of wastage rate of 2 1/2% Cr-1 Mo steel was obtained, including parameters of sodium temperature, steam injection rate, and nozzle-to-target distance. The depth curves and the widths of the area affected by wastage were examined. These test results were applied to the safety considerations for the prototype steam generators in Japan.

- 3431 SUBCRITICALITY GUIDELINES FOR LIQUID-METAL FAST BREEDER REACTOR SPENT FUEL SHIPPING

CASK DESIGNS. Philbin, J.S.; Dupree, S.A. (Sandia Labst, Albuquerque, NM). Nucl. Technol.; 40: No. 3, 284-296(Oct 1978).

A series of three-dimensional Monte Carlo criticality calculations has been performed to examine the criticality problems that exist in liquid-metal fast breeder reactor spent fuel shipping casks. The reactivity effects of subassembly loading sequences, total fuel payloads, subassembly pitches, neutron poisons, and choices of cask coolants and structural materials were considered. The analyses showed that, for a given type of fuel, the system k /sub eff/ can be displayed in a manner that permits determination of the maximum number of subassemblies that can be safely loaded in a particular cask. For a given cask coolant, this maximum number can be fixed independent of (1) the sequence or positions in which the subassemblies are loaded; (2) the detailed design of the basket supporting the subassemblies, including small changes in the distance between the subassemblies; and (3) changes in the cask design outside the fuel region. Thus, the shielding, structural, and thermal problems inherent in the design of the outer cask structure can be addressed without reference to the criticality problem.

3432 THEORY OF VOID NUCLEATION IN METALS. Russell, K.C. (Massachusetts Inst. of Tech., Cambridge). Acta Metall.; 26: No. 10, 1615-1630(Oct 1978).

A general theory of void nucleation in irradiated metals is developed via the nodal line/critical point formalism of Poincare. The theory reduces to several simple subcases, depending on the experimental conditions, in particular the damage rate, temperature, and gaseous and non-gaseous impurity concentrations. The most important of these subcases are: spontaneous (i.e., no activation barrier) void nucleation driven by inert gas, heterogeneous nucleation on metastable gas: vacancy clusters, heterogeneous nucleation on clusters of non-gaseous impurities, and homogeneous nucleation with or without the assistance of mobile, surface active impurities. Simple nucleation rate equations are presented for each of these cases. Equations for void number density are derived for the various steady state nucleation cases on the basis of a simple void growth equation. The predictions of void number densities are found in reasonable agreement with measurements on pure Ni, type 316 stainless steel, and on the Nimonic alloy, PE-16. The theory also reproduces the shape of the void number density vs. $1/T$ curves, including the activation energies.

3433 CORE COMPONENT MATERIALS FOR SODIUM-COOLED FAST REACTORS. Harries, D.R. (UKAEA, Harwell, Engl). J. Inst. Nucl. Eng.; 17: No. 4, 301-311(Oct 1978).

The principal technological consequences of irradiation-induced void swelling and creep in the materials already chosen or being considered for core component (particularly the fuel element subassembly wrapper or duct) applications in sodium-cooled fast reactors are summarized. The current understanding of the mechanisms of the phenomena is briefly described and the influences of irradiation and materials parameters in determining the magnitudes of the void swelling and irradiation creep are reviewed.

3434 CORROSION OF MATERIALS FOR SODIUM-COOLED FAST REACTORS. Antill, J.E. (UKAEA, Harwell, Engl). J. Inst. Nucl. Eng.; 17: No.

4, 313-319(Oct 1978).

Major compatibility topics in sodium-cooled fast reactors are considered and cover the corrosion of boiler tube materials by aqueous environments in boilers and by the products of sodium-water reactions and the interaction of structural materials with sodium coolants. Reference is also made to the cleaning of surfaces and the tribology of sliding interfaces.

3435 EXPERIENCE IN SODIUM TECHNOLOGY OF SOVIET FAST REACTORS. Karpov, A.V.; Archipov, V.M.; Archangel'skii, V.V.; Ivanenko, V.N. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). Kernenergie; 20: No. 10, 322-326(Oct 1978). (In German).

From CMEA specialist meeting on problems of technology and corrosion in sodium coolant and cover gas; Dresden, German Democratic Republic (28 Mar - 1 Apr 1977).

Experience gathered with sodium technology during operation of the Soviet fast reactors BR-5, BOR-60 and BN-350 is considered. Problems on pretreatment of the sodium coolant before filling the main circuits of the reactors are briefly presented. The results of studies on impurities in the coolant during start-up operations and during operation are discussed. Finally, some problems of equipment cleaning, sampling, and fighting against sodium fires are considered. Some experience in cover gas technology is given.

3436 COMPATIBILITY OF STAINLESS-STEEL CLADDING MATERIALS WITH URANIUM CARBIDE FUEL. Stumpf, W.E.; Venter, S. (Atomic Energy Board, Palindaba, Pretoria (South Africa). Physical Metallurgy Div.). J. S.Afr. Inst. Min. Metall.; 79: No. 3, 51-58(Oct 1978).

Some of the factors that govern the choice of a nuclear fuel are briefly discussed, special reference being made to the introduction of advanced carbide fuels into fast-breeder programmes. One of these factors, i.e. the compatibility of carbide fuels with stainless-steel cladding, is discussed in depth. A very sensitive technique for the study of compatibility interactions between fuel and cladding is described in respect to the effects of particle size, specific surface area and surface oxidation of the fuel, and microstructure of the cladding on the compatibility behaviour of the cladding with the fuels. Finally, the need for the development of modified or stabilized uranium carbide fuels is discussed, and some results are presented to show the effects of small amounts of sulphur in these fuels on their compatibility with stainless-steel cladding materials.

3437 CORE COMPONENT MATERIALS FOR SODIUM-COOLED FAST REACTORS. Harries, D.R. (UKAEA, Harwell. Atomic Energy Research Establishment). Nucl. Energy; 17: No. 4, 301-311(Oct 1978).

From INCO power conference 1977; Lausanne, Switzerland (5 - 7 Oct 1977).

The principal technological consequences of irradiation-induced void swelling and creep in the materials already chosen or being considered for core component applications, particularly the fuel element subassembly wrapper or duct, in sodium-cooled fast reactors are summarized. The current understanding of the mechanisms of the phenomena is briefly described and the influences of irradiation and materials parameters in determining the magnitudes of the void swelling and irradiation creep are reviewed.

- 3438 CORROSION OF MATERIALS FOR SODIUM-COOLED FAST REACTORS. Antill, J.E. (UKAEA, Harwell. Atomic Energy Research Establishment). Nucl. Energy; 17: No. 4, 313-319 (Oct 1978).

From INCO power conference 1977; Lausanne, Switzerland (5 - 7 Oct 1977).

Major compatibility topics in sodium-cooled fast reactors are considered. They cover the corrosion of boiler tube materials by aqueous environments in boilers and by the products of sodium-water reactions, and the interactions of structural materials with sodium coolants. Reference is also made to the cleaning of surfaces and the tribology of sliding interfaces.

- 3439 STATUS OF FRENCH BREEDER DEVELOPMENT AND GERMAN-FRENCH COOPERATION. Vendryes, G. (CEA, 75 - Paris (France). Direction des Applications Industrielles Nucleaires); Hennies, H.H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). Atomwirtsch., Atomtech.; 23: No. 10, 448-451 (Oct 1978). (In German).

The development of fast breeder reactors in France is at present characterized by the successful operation of the 250 MW reactor, Phenix, and the speedy progress made in construction of the 1200 MW Superphenix plant. The technical concept of Superphenix is a logical extension of the Phenix concept, i.e., the pool type sodium cooled breeder reactor. While the only change in the primary system has been an increase in capacity, the main modification over Phenix is the change from small modular steam generators to four large units. The development of the work to date does not cast any doubt upon the envisaged date of first criticality in the second half of 1982. The basis of German-French cooperation is a joint declaration by the then Ministers for Research of the two countries on February 13, 1976 in which the Ministers had expressed themselves in favor of close cooperation between both countries in the development of safe, reliable and economic fast breeder reactors. On this basis, the two governments, the research centers and the industrial partners in the two countries, including the previous partners in Belgium, the Netherlands and Italy, agreed on the general principles of this cooperation. Meanwhile, the cooperation formalized in 1977 has been activated. Both in research and development and among planning and building industries the exchange of know-how and cooperation is in full swing. The joint company, SERENA, is the pool for the know-how introduced by the partners and the sole representative of the partners in the scheme. (orig.) 691 UA.

- 3440 BREEDER DEVELOPMENT AND FRANCO-GERMAN COOPERATION. Vendryes, G.; Hennies, H.H. (CEA, Paris, Fr.). Atomwirtsch., Atomtech.; 23: No. 10, 448-451 (Oct 1978). (In German).

The development of fast breeder reactors in France is at present characterized by the successful operation of the 250-MW reactor, Phenix, and the speedy progress made in construction of the 1200-MW Superphenix plant. The technical concept of Superphenix is a logical extension of the Phenix concept, i.e., the pool-type sodium-cooled breeder reactor. While the only change in the primary system has been an increase in capacity, the main modification over Phenix is the change from small modular steam generators to four large units. The development of the work to date does not cast any doubt upon the envisaged date of first criticality in the second half of 1982. The basis of German-French cooperation is a

joint declaration by the then Ministers for Research of the two countries on February 13, 1976 in which the Ministers had expressed themselves in favor of close cooperation between both countries in the development of safe, reliable and economic fast breeder reactors. On this basis, the two governments, the research centers and the industrial partners in the two countries, including the previous partners in Belgium, the Netherlands and Italy, agreed on the general principles of this cooperation. Meanwhile, the cooperation formalized in 1977 has been activated. Both in research and development and among planning and building industries the exchange of know-how and cooperation is in full swing. The joint company, SERENA, is the pool for the know-how introduced by the partners and the sole representative of the partners in the scheme.

- 3441 ELECTRON BEAM WELDING OF AUSTENITIC STAINLESS STEEL. ELEVATED TEMPERATURE PROPERTIES. Susei, S.; Matsui, S.; Nagai, H.; Mori, E. (Kawasaki Heavy Industries Ltd., Akashi, Hyogo (Japan). Technical Inst.); Shimizu, S. Kawasaki Gihō; No. 68, 20-27 (Oct 1978). (In Japanese).

Austenitic stainless steel is used for liquid metal-cooled fast breeder reactors with operating temperature of about 550 deg C, because its elevated temperature properties are excellent and the results of use are abundant. The welded joints in LMFBRs require high degree of safety, and the application of electron beam welding is studied to make welding joints of high quality. When the inelastic deformation in a certain limit is allowed as prescribed in the ASME Code, Case 1592, the elevated temperature properties of the welded joints of structures are particularly important. The materials tested were 10 mm thick plates of SUS 304, SUS 316 and SUS 321 steels, and 150 kV - 40 mA electron beam welder was employed. The method of welding was one side, one pass Uranium welding, and first, the appropriate welding conditions were decided. Elevated temperature tensile test was carried out on the parent materials and welded joints by electron beam welding and coated arc welding. Creep rupture test and elevated temperature fatigue test were also carried out. In EB-welded austenitic stainless steel, delta ferrite is scattered finely in austenite, and its quantity is very small and less than 1.5%. The tensile strength and 0.2% proof stress of EB-welded joints are almost same as those of parent materials. The creep rupture and fatigue properties of the joints are also close to those of parent materials.

- 3442 COMPILATION OF REFERENCES, DATA SOURCES AND ANALYSIS METHODS FOR LMFBR PRIMARY PIPING SYSTEM COMPONENTS. Reich, M.; Esztergar, E.P.; Ellison, E.G.; Erdogan, F.; Gray, T.G.F.; Wells, C.H. (Brookhaven National Lab., Upton, NY (USA)). Nucl. Eng. Des.; 50: No. 2, 273-304 (Oct 1978).

A survey and review program for application of fracture mechanics methods in elevated temperature design and safety analysis was initiated in December of 1976 by the Structural Analysis Group at Brookhaven National Laboratory. The aim of this paper is to provide a critical review of the theories of fracture and the application of fracture mechanics methods to life prediction, reliability and safety analysis of piping components in nuclear plants undergoing sub-creep and elevated temperature service conditions.

- 3443 RECENT RESEARCHES ON THERMAL-

HYDRODYNAMIC TRANSIENT PHENOMENA EXPERIENCED IN LMFBRs. Maekawa, I.; Koishikawa, S.; Sakano, K. (Kawasaki Heavy Industries Ltd., Kobe (Japan)). FAPIG (Tokyo); No. 90, 11-19 (Oct 1978). (In Japanese).

Recent researches on thermal-hydrodynamic transient phenomena and the hydrodynamic load due to the phenomena are introduced in this paper. Investigations were made on the items of the drastic transient phenomena due to the break down of the pipings of steam-water system, the propagation of pressure waves in sodium pipings at the time of sodium-water reaction, and the response of pipings to hydrodynamic loading. The thermodynamical non-equilibrium model shows good agreement with the experimental results regarding the propagation of reduced-pressure waves. When the pressure loss can be neglected, initial critical flow rate is smaller than steady critical flow rate. The results of analytical studies on the propagation of pressure waves were compared with the experimental results. The cavitation in sodium was in good agreement with the liquid column separation model. The analysis of the propagation of pressure waves due to the elastoplastic response of pipings was in good agreement with experimental results by the consideration of plastic deformation effect. The hydrodynamic loading following the blowdown of water-steam system and sodium flow-in reaches to several hundreds of tons. The stress of pipings due to the hydrodynamic loading is larger than that due to the own weight of the pipings or earthquake loading.

3444 METHOD AND APPARATUS FOR PRE-HEATING DUCTS FORMING LIQUID SODIUM FLOW PASSAGE. Nanba, M. (to Toshiba Corp., Kawasaki, Kanagawa (Japan)). Japanese Patent 1953-11,0797/A/. 27 Sep 1978. 3p. (In Japanese).

A method to pre-heat ducts by connecting an AC power source to the exciting coil of a flowmeter provided in the duct for the measurement of a sodium flow rate and exerting the magnetic field resulted by the coil on the duct is described. Upon charging liquid sodium in a non-magnetic duct such as made of stainless steel for forming flow passage for the sodium, AC voltage is applied through a turn-over switch to the excitation coil of an electromagnetic flowmeter provided to the duct. The flowmeter thus operates as an induction type heater to produce eddy current in the duct and pre-heat the same. Upon turning-over the above switch to apply DC voltage to the excitation coil, the flowmeter operates as a flow-measuring instrument.

3445 LIQUID METAL COOLED NUCLEAR REACTOR. Marcotte, J.L.B.; Chevalier, G.; Gerij, J.; Leheu, J. (to Commissariat a l'Energie Atomique). US Patent 4,116,765. 26 Sep 1978. Filed date 16 Dec 1976. 6p.

The concrete vault roof of a liquid metal cooled fast reactor is provided with a heat-insulating lining plate and has a central passage for rotating shield plugs surrounded by passages for heat exchangers and pumps. An annular plate of relatively small thickness is placed horizontally beneath the vault roof and its external edge is joined to the lining plate by means of shaped members having a long radius of curvature so as to form closed spaces for trapping an inert gas. The free edges formed by the openings in the annular plate are joined to the lining plate by means of a sealing shell.

3446 ULTRASONIC DIP SEAL MAINTENANCE SYSTEM. Poindexter, A.M.; Ricks, H.E. (to Dept. of

Energy). US Patent 4,116,766. 26 Sep 1978. Filed date 31 Aug 1976. 10p.

PAT-APPL-719,310.

Disclosed is a system for removing impurities from the surfaces of liquid dip seals and for wetting the metal surfaces of liquid dip seals in nuclear components. The system comprises an ultrasonic transducer that transmits ultrasonic vibrations along an ultrasonic probe to the metal and liquid surfaces of the dip seal thereby loosening and removing those impurities.

3447 SAFETY APPLIANCE FOR A STEAM GENERATOR HEATED BY LIQUID METAL. Taylor, D.; Watts, M.J. (to UKAEA London Office; Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 1,936,409/C/. 21 Sep 1978. 4p. (In German).

The invention should increase the safety of a heat exchanger (sodium/water). It is characterized by a safety tube connected to the lower end of the vessel that contains the tubes for both liquids, which leads to a catch tank installed below the vessel. If a heatpipe of the vessel breaks and the pressure increases because of the Na-H₂O reaction, the rupture disk in the safety pipe responds and the sodium is extracted from the reaction zone by pressure and gravity and flows into the catch tank. The invention is explained by the example of a steam generator in a sodium cooled fast breeder reactor.

3448 FAST NEUTRON NUCLEAR REACTOR. Jogand, P. (to Commissariat a l'Energie Atomique). US Patent 4,115,192. 19 Sep 1978. Priority date 6 Jul 1976, France, 4p.

A fast neutron nuclear reactor is described comprising on the inside a main vessel having a vertical axis suspended by its upper open part on a thick protective slab which seals a cavity surrounding said vessel, a volume of a liquid cooling metal and also positioned within said main vessel an inner vessel having a cylindrical wall containing the core of the reactor submerged beneath the liquid metal level, the intermediate space between the main vessel and the inner vessel having heat exchangers and pumps which pass below the level of the liquid metal, wherein the upper end of the inner vessel is covered by an inverted annular bell cap and is placed under the internal pressure of a neutral gas in order to prevent in normal operation direct communication between the hot liquid metal within the inner vessel and the cold liquid metal in the annular space between the main vessel and the inner vessel, said bell cap covering an auxiliary exchanger which externally surrounds the inner vessel in such a way that if there is a fault in the reactor and after reducing the neutral gas pressure beneath the bell cap as a result of natural convection a circulation of the liquid metal takes place beneath the bell cap of the inner vessel towards the annular space and passes through the auxiliary exchanger.

3449 NUCLEAR REACTOR PRESSURE VESSEL SUPPORT SYSTEM. Sepelak, G.R. (to Dept. of Energy). US Patent 4,115,193. 19 Sep 1978. Filed date 17 Aug 1976. 10p. PAT-APPL-715,255.

Disclosed is a support system for nuclear reactor pressure vessels which can withstand all possible combinations of stresses caused by a postulated core disrupting accident during reactor operation. The nuclear reactor pressure vessel is provided with a flange around the upper periphery thereof, and the flange

includes an annular vertical extension formed integral therewith. A support ring is positioned atop of the support ledge and the flange vertical extension, and is bolted to both members. The plug riser is secured to the flange vertical extension and to the top of a radially outwardly extension of the rotatable plug. This system eliminates one joint through which fluids contained in the vessel could escape by making the fluid flow path through the joint between the flange and the support ring follow the same path through which fluid could escape through the plug risers. In this manner, the sealing means to prohibit the escape of contained fluids through the plug risers can also prohibit the escape of contained fluid through the securing joint.

3450 CONTROL OF NUCLEAR POWER POSES CHALLENGE TO U.S. Freeman, S.D. Energy User News; 3: No. 38, 22-23(18 Sep 1978).

Excerpted from remarks made at the Atomic Industrial Forum Fuel Cycle Conference held in New York City.

S. David Freeman, Chairman, the Tennessee Valley Authority, criticizes the nuclear industry for what he considers childish reactions to competition from newly funded energy alternatives when the bulk of research and development funds continue to go for nuclear power. Freeman favors the use of both nuclear and coal-fired power plants on the grounds that neither option can supply the entire demand without serious environmental and security problems. He considers solar energy to be technologically and economically feasible for some applications and a serious competitor to breeder and fusion reactors in the future. He sees time as the real enemy in resolving the energy crisis and emphasizes the importance of having the proponents of specific energy sources work together to resolve the conflicts that are preventing progress. He would resolve the breeder debate by (1) using a fuel cycle that discourages proliferation, (2) incorporates the known state-of-the-art, and (3) is funded equally by the public and the utilities.

3451 (DOE-tr--61) DIRECT NUMERICAL SIMULATION OF TURBULENT VELOCITY, PRESSURE, AND TEMPERATURE FIELDS IN CHANNEL FLOWS.

Groetzsch, G. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). 15 Sep 1978. Translation of KFK--2426, October 1977 217p. Dep. NTIS (US Sales Only), PC A10/MF A01.

Work performed under United States--W. Germany Fast Reactor Exchange Program.

For the simulation of nonsteady, three-dimensional, turbulent flow- and temperature-fields in channel flows of fluids with constant properties a method is presented which is based on a finite difference scheme of the complete conservation equations for mass, momentum, and enthalpy. The fluxes of momentum and heat within the grid cells are described by subgrid scale models. The subgrid scale model for momentum introduced here is for the first time applicable to small Reynolds numbers, rather coarse grids, and channels with space-dependent roughness distributions. The new developed subgrid scale model for temperature also takes the influence of the molecular Prandtl number into consideration. For Prandtl numbers of about one, the theoretical model confirms the empirically derived values for the turbulent subgrid scale Prandtl number, which are currently used by other authors. For very small molecular Prandtl numbers of liquid metals, the model shows that the total resolution of all relevant temperature fluctuations can be

achieved on the basis of presently realizable grids. The method is realized in the computer code TURBIT-2. It can be used for the simulation of laminar flows in plane channels and annuli as well as for turbulent flows with Reynolds numbers larger than about 10^4 . The numerical results for the flow field show only a small influence of the model parameters. For smooth, rough, and partly roughened channels with secondary flows the agreement with experimental results is fairly good even when rather coarse grids are used. The temperature fields simulated are very insensitive to the parameters of the model. The numerical results are in good agreement with consistent, reliable experimental results.

3452 NUCLEAR REACTOR WITH EQUIPMENT FOR PREVENTING THE CONVECTION OF METAL VAPORS. Lakra, R.J.; Freund, J.; Rauschert, G. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.); Deutsches Patentamt, Muenchen (Germany, F.R.)). German(FRG) Patent 2,709,914/A/. 14 Sep 1978. 11p. (In German).

Convection of Na-vapors and -aerosols towards cooler parts of the reactor above the coolant surface and condensation of the metal on these parts is prevented. This is achieved by means of quasi-tight convection barriers in the form of e.g. conical sheets arranged in the gas plenum and dividing it with respect to its height. If the closure head consists of several rotating shields embossed in each other the convection barriers are constructed out of several single sheets and designed congruently and coaxially rotatable with the shields. A gas possibly existing between reactor vessel and closure head is protected by means of a labyrinth packing.

3453 CORE DISRUPTIVE ACCIDENT MARGIN SEAL. Garin, J.; Belsick, J.C. (to Dept. of Energy). US Patent 4,113,564. 12 Sep 1978. Filed date 13 Aug 1976. 8p. PAT-APPL-714,220.

Disclosed is an apparatus for sealing the annulus defined between a substantially cylindrical rotatable first riser assembly and plug combination disposed in a substantially cylindrical second riser assembly and plug combination of a nuclear reactor system. The apparatus comprises a flexible member disposed between the first and second riser components and attached to a metal member which is attached to an actuating mechanism. When the actuating mechanism is not actuated, the flexible member does not contact the riser components thus allowing the free rotation of the riser components. When desired, the actuating mechanism causes the flexible member to contact the first and second riser components in a manner to block the annulus defined between the riser components, thereby sealing the annulus between the riser components.

3454 APPARATUS FOR CONTROLLING COOLANT LEVEL IN A LIQUID-METAL-COOLED NUCLEAR REACTOR. Jones, R.D. (to Dept. of Energy). US Patent 4,113,562. 12 Sep 1978. Filed date 20 Apr 1977. 6p. PAT-APPL-789,154.

A liquid-metal-cooled fast-breeder reactor which has a thermal liner spaced inwardly of the pressure vessel and includes means for passing bypass coolant through the annulus between the thermal liner and the pressure vessel to insulate the pressure vessel from hot outlet coolant includes control ports in the thermal liner a short distance below the normal

operating coolant level in the reactor and an overflow nozzle in the pressure vessel below the control ports connected to an overflow line including a portion at an elevation such that overflow coolant flow is established when the coolant level in the reactor is above the top of the coolant ports. When no makeup coolant is added, bypass flow is inwardly through the control ports and there is no overflow; when makeup coolant is being added, coolant flow through the overflow line will maintain the coolant level.

- 3455 NUCLEAR REACTOR CORE CONFIGURATION. Baxi, C.B. (to General Atomic Co., San Diego, CA (USA); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,808,370/A/. 7 Sep 1978. 19p. (In German).
- The high temperature reactor cooled by He (or also a reactor cooled by Na) has got a core of hexagonal geometry with fuel elements and, arranged around them in three rows, breeding blanket elements. Both types of elements have got flow guiding devices. In addition, in each breeding blanket element there is provided, looking in flow direction, before or behind the breeding blanket rods, arranged in compact packing in the element, a flow restrictor. It consists of an inner casing in which are arranged rods, aligned parallel to each other and to the flow direction, with a roughened surface. The flow restrictor makes use of the principle that the friction factor is essentially reduced if the Reynolds number of the flow changes from e.g. 4×10^3 to 2×10^3 . This causes the percentage of the total coolant flow passing through the breeding blanket elements to become larger at low circulation rate than that of the total coolant flow at full load operation.

- 3456 COLLECTING TANK WITH EVAPORATION PROTECTION FOR A NUCLEAR REACTOR. Pischel, G.; Dumm, K.; Friedrich, H.J. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,709,471/A/. 7 Sep 1978. 11p. (In German).
- In order to prevent evaporation and aerosol formation in the containment e.g. of a FBR in case of the coolant (e.g. sodium or fused salt) escaping into it after a Bethe-Tait accident, floating bodies are placed in the collecting tank. These are axially symmetrical, especially spherical, hollow bodies made of e.g. chrome-nickel steel. They are present in the collecting tank in such number that in beginning to swim they completely cover the surface of the coolant. In order to prevent tumbler or rotation movements causing each time fresh coolant to be brought over the layer of floating bodies, which then could evaporate, these floating bodies are provided with ballast weights keeping them in a well-defined position. For depressurization of the interior of the floating bodies, these have got a relief boring in the region of the ballast weights. For utilization of the maximum degree of covering the coolant surface the floating bodies sink into the coolant with half their volume.

- 3457 PACKED ROD NEUTRON SHIELD FOR FAST NUCLEAR REACTORS. Eck, J.E.; Kasberg, A.H. (to Dept. of Energy). US Patent 4,111,747. 5 Sep 1978. Filed date 8 Jun 1976. 8p. PAT-APPL-693,990.
- Disclosed is a fast neutron nuclear reactor including a core and a plurality of vertically

oriented neutron shield assemblies surrounding the core. Each assembly includes closely packed cylindrical rods within a polygonal metallic duct. The shield assemblies are less susceptible to thermal stresses and are less massive than solid shield assemblies, and are cooled by liquid coolant flow through interstices among the rods and duct.

- 3458 EFFECT OF SWELLING AND CREEP ON LIQUID-METAL FAST BREEDER REACTOR PERFORMANCE. Barthold, W.P.; Beitel, J.C. (Argonne National Lab., IL). Nucl. Technol.; 40: No. 2, 138-148 (Sep 1978).

The impact of swelling and creep phenomena, as expressed in the various Nuclear Systems Materials Handbook correlations and their updates, on liquid-metal fast breeder reactor (LMFBR) fuel assembly duct design and LMFBR performance was investigated. As more data on irradiation-induced creep and swelling became available, the predicted creep and swelling rates increased. This leads to increases in interassembly gaps required for the same duct life. Penalties in breeding ratio, doubling times, and fuel cycle cost are the result of the increased interassembly gaps. The most significant penalties are encountered when the updated correlations on creep and swelling are used to update the duct life calculation. To compensate for increases in predicted duct dilation, the duct lifetime has to be reduced significantly.

- 3459 OPERATION OF THE WATER-TO-SODIUM LEAK DETECTION SYSTEM AT THE EXPERIMENTAL BREEDER REACTOR II. Osterhout, M.M. (Argonne National Lab., IL). Nucl. Technol.; 40: No. 2, 159-169 (Sep 1978).

A water-to-sodium leak detection system was installed at the Experimental Breeder Reactor II in April 1975. The system is designed for early detection of steam generator leaks, using hydrogen meters at the sodium outlets of the evaporators and superheaters. The leak detectors operate by measuring the rate of diffusion of hydrogen from the liquid sodium through a nickel membrane into a dynamic vacuum system. The advantages of this detection system are rapid response time, high sensitivity, stability, and reliability. The system was operated on an experimental basis for the first two years. During this period, data were obtained on detector stability, reliability, maintenance needs, computer interface requirements, calibration, and background hydrogen-level fluctuations. A generic defect in the original detectors was also discovered, requiring redesign of the units. When the new units were installed and proven to be reliable, the system was made fully operational. The data from the hydrogen meters are now used as the primary basis for detection of water-to-sodium leaks.

- 3460 (JAPFNR--336) PROGRESS REPORT ON FAST BREEDER REACTOR DEVELOPMENT IN JAPAN, JANUARY--MARCH 1978. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Sep 1978. Translation of PNC-N--251-78-03 13p. Dep. NTIS (US Sales Only), PC A02/MF A01.
- Work performed under United States--Japanese Fast Reactor Exchange Program.

Information is presented concerning the development of the Joyo Reactor; development of the Monju Reactor; reactor physics; development of reactor components; instrumentation and control; sodium technology; fuel materials; structural materials; reactor safety; and steam generators.

- 3461 BREEDER--NOW OR NEVER. Murphy, P.M. (EPRI, Palo Alto, Calif.). Mech. Eng.; 100: No. 9, 22-24(Sep 1978).

The timing of the commercial introduction of the LMFBR (Liquid Metal Fast Breeder Reactor) will be an important factor in its ability to supply a significant fraction of the nation's future electrical needs. The number of breeders we can build initially will be limited by the size of our low-cost uranium resources and by the rate at which LWR's (Light Water Reactors) are placed in service. Since this uranium resource is fixed in size while electrical demand will grow geometrically, it is clear that the sooner the breeder is introduced commercially the larger will be the fraction of electrical demand that it can supply. An early commercial introduction on an adequate scale requires full-scale resumption of LWR construction and redirection of LMFBR development programs toward a near-term commercial prototype.

- 3462 FAST BREEDER REACTOR I. Klose, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). Phys. Bl.; 34: No. 9, 407-419(Sep 1978). (In German).

The paper deals with the neutron balance in the chain reaction in LWR reactors and fast breeders and with the role of delayed neutrons. Safety problems are discussed in great detail. A short view on the SBR prototype SNR 300 now under construction is followed by a comparison of the different reactor types. Finally, some political problems and the proliferation problem are briefly commented upon.

- 3463 ^{235}U , ^{238}U , AND ^{239}Pu α VALUE MEASUREMENTS IN THE BOR-60 REACTOR CORE. Prokopenko, V.S.; Gabeskiya, V.Ya.; Inchagov, A.V.; Novikov, Yu.B.; Prokop'ev, V.M.; Tikhomirov, V.V.; Chetverikov, A.P. At. Energ. (USSR); 45: No. 3, 230-231(Sep 1978). (In Russian).
BOR-60 REACTOR; CAPTURE-TO-FISSION RATIO; DATA; ERRORS; NEUTRON REACTIONS; PLUTONIUM 239 TARGET; REACTOR CORES; URANIUM 235 TARGET; URANIUM 238 TARGET

- 3464 INVESTIGATION OF THE BOR-60 REACTOR DYNAMIC CHARACTERISTICS WITH A REACTIVITY OSCILLATOR. Afanas'ev, V.A.; Britov, V.N.; Gryazev, V.M.; Efimov, V.N.; Kebabze, B.V.; Krauze, Kh.; Otto, Kh.; Shol'ts, J.; Tsigenbaja, D. At. Energ. (USSR); 45: No. 3, 175-179(Sep 1978). (In Russian).

The BOR-60 reactor power coefficient GITA and its time constant τ have been investigated. The reactivity oscillator represents a rotating cylinder, wherein d neutron absorber B_4C and a scatterer $\text{Zrsub}(2)\text{Hsub}(1.82)$ are located on opposite sides. The shape of the periodic reactivity disturbance is close to the harmonic one with a double amplitude approximately $2 \times 10^{-4} \Delta k/k$. The bench-mark and in-pile oscillator experimental results are presented. The on-line pile oscillation technique is shortly described. The amplitude and phase-frequency characteristics have been measured at reactor power $N=22$ and 45 MW in the frequency range from 0.05 to 18 Hz. The results have proved the applicability of a simple model with one GITA and related τ for the reactor dynamics at the power regimes investigated and $f > 0.5$ Hz.

- 3465 (DOE-tr--90) BEHAVIOUR OF THE CORE OF A FAST SODIUM-COOLED BREEDER REACTOR OF 2000 MW(E) DURING DISRUPTIVE ACCIDENTS WITH A VERY

LOW LIKELIHOOD OF OCCURRENCE. Struwe, D. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). Inst. fuer Neutronenphysik und Reaktortechnik. Sep 1978. Translation of KFK--2490 204p. Dep. NTIS (US Sales Only), PC A10/MF A01.

Work performed under United States--W. Germany Fast Reactor Exchange Program.

A structural phase diagram for hypothetical core disruptive accidents (HCDA) has been developed based on a variety of analyses for different LMFBRs. The intent was to identify the strategic phases of HCDAs which are important with regard to the safety aspects of the plant. These phases are investigated in detail for a 2000 MW(e) LMFBR (SNR-2000). Characteristic data of the SNR-2000 are discussed concerning their effect on safety analysis. Reasons for the choice of model parameters for special phenomena such as fuel-coolant interaction, fuel rod failure mechanisms, and sodium boiling are given. The results of calculations with CAPRI-2, HOPE and KADIS are analyzed for possibilities of entering into energetic core disassembly with consequences which would require power values below 2000 MW(e).

- 3466 BUBBLE NOISE AT MAGNETIC FLOWMETERS. Fuge, R.; Langenbrunner, H. (Zentralinstitut fuer Kernforschung, Rossendorf bei Dresden (German Democratic Republic)). Nucl. Eng. Des.; 49: No. 3, 205-212(Sep 1978).

The effect of bubbles in the liquid flow on the output signal of magnetic flowmeters is theoretically and experimentally investigated. In the theoretical model the bubble noise is due to the superposition of single pulses generated by individual bubbles which are randomly distributed in the liquid. On the basis of this model the rms value of the bubble noise is found to be proportional to the void fraction, and inversely proportional to the square root of the bubble flowrate. Owing to electromagnetic forces the factor of proportionality depends on the imposed magnetic field. The theoretical results are verified by experiments in sodium test loops.

- 3467 LONG-TERM ATOMIC ENERGY RESEARCH, DEVELOPMENT AND UTILIZATION PROGRAM FORMULATED. At. Jpn.; 22: No. 9, 5-19(Sep 1978).

Japan Atomic Energy Commission has submitted the report on the new 'Long-term Atomic Energy Research, Development and Utilization Program' to the Government. It reflects recent severe circumstances represented by the oil crisis, opposition to nuclear power, and the nuclear non-proliferation issue. Although it covers very wide range of atomic energy industry, the basis of this report can be summarized as follows. (1) To improve the reliability and availability of LWRs, and implement standardization. (2) Commercialization of FBR by the year 2000. (3) Practical application of nuclear fusion at the earliest possible date in the 21st Century. Time schedule and necessary funds are estimated for each specific item of research and development projects.

- 3468 PERFORMANCE ANALYSIS OF MESH-PACKED COLD TRAPS. Murase, M.; Sumida, I.; Kotani, K.; Yamamoto, H. (Hitachi Ltd., Ibaraki (Japan). Energy Research Lab.). J. Nucl. Sci. Technol. (Tokyo); 15: No. 9, 711-713(Sep 1978).

LMFBR. COLD TRAPS; COOLANT CLEANUP SYSTEMS; COOLANTS; EFFICIENCY; LIQUIDS; PERFORMANCE; PURIFICATION; SODIUM; SODIUM OXIDES; LMFBR TYPE REACTORS

3469 APPLICATION OF A MICROPROCESSOR-BASED DISTRIBUTED SYSTEM TO SUBASSEMBLY FAULT DETECTION OF LMFB. Kimura, Y.; Hasegawa, K.; Sekiguchi, A. (Tokyo Univ. (Japan). Faculty of Engineering). Tokyo Daigaku Kogakubu Fuzoku Sogo Shikenjo Nenpo; 37: 167-172(Sep 1978). (In Japanese).

Single subassembly fault detection of LMFB requires the use of individual subassembly instrumentation and the processing of outgoing signals. A computerized system instead of a conventional hardwired electronic one will likely have to be used for handling a large number of these signals. There are, however, some problems in using a computerized system in high reliability application. The adoption of a microprocessor-based distributed system will be one of the means to solve these problems. This report describes the performance estimation of a proposed microprocessor-based distributed subassembly fault detection system. The results show the feasibility of the proposed system.

3470 DEVELOPMENT OF COLD TRAP FOR LMFB. Ohshima, I.; Ohtani, R.; Nei, H. (Toshiba Corp., Kawasaki, Kanagawa (Japan). Research and Development Center); Adachi, S.; Honda, M. Toshiba Rebyu; 33: No. 9, 782-786(Sep 1978). (In Japanese).

The purity control for the liquid sodium is an important task for reliable operation of sodium-cooled fast breeder reactors. In view of early detection of water leakage in the steam generator, detectors of the hydrogen concentration monitoring type are equipped. For satisfactory function of the detectors, it is necessary that the hydrogen concentration is kept below a definite low level. The writers, abstracting the problematic points about the cold trap in the secondary cooling system, developed a design technique to solve the problems, carried out experiments on the hydrogen removing function of cold trap, and made analytical evaluation to confirm the validity of the design technique. The reliability of the cold traps and, consequently, of fast breeder reactor plants is expected to be raised by this achievement.

3471 FRACTURE OF NUCLEAR PIPING CAUSED BY IMPACT LOADING. Yagawa, G.; Aizawa, T.; Ando, Y. (Tokyo Univ. (Japan). Faculty of Engineering). Tokyo Daigaku Kogakubu Fuzoku Sogo Shikenjo Nenpo; 37: 181-186(Sep 1978). (In Japanese).

With the modelled specimen of the primary piping for a nuclear reactor, the experiment of dynamic rupture was performed to clarify the dynamic behavior of a pipe with a crack. Also, the effect of creep damage to the dynamic rupture was evaluated using plate test pieces which were subjected to various degrees of creep damage before dynamic test. The special experimental apparatus which gives mechanical shock to test specimens with the solenoid and the lever mechanism, the mechanical properties of each specimen, the shapes and dimensions of the specimens for creep, uniaxial tensile and impact loading test, the instrumentation system, the conditions of creep test and the three types of initial crack in the modelled specimens are presented. As for the experimental results and the relating evaluation, the nondimensional parameters under plane stress condition being introduced by Hutchinson's and the author's methods are compared, and the characteristic stress distribution at the crack tip is reported. These evaluations were conducted under plastic deformation. The time response of net stress and axial strain in the test specimens without notch and with an inner circumferential notch

are shown. From the test results and the proposed evaluation method, it was found that the nonlinear fracture mechanics is available for the reasonable design of structures under elevated temperature.

3472 (DOE-tr--40) FAST BREEDER PROJECT. SECOND QUARTERLY REPORT, 1977. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). 31 Aug 1978. Translation of KFK--1277/2, October 1977 209p. Dep. NTIS (US Sales Only), PC A10/MF A01.

Work performed under United States--W. Germany Fast Reactor Exchange Program.

Progress is summarized in the areas of fuel rod development, materials studies, reactor kinetics, reactor safety, and cooling system technology.

3473 TUBE BOILER HEAT EXCHANGER. Hayden, D.; Sheward, G.E.; Ormrod, D.T.; Willby, C.R.; Firth, G.F. (to UKAEA; Deutsches Patentamt, Muenchen (Germany, F.R.)). German(FRG) Patent 2,705,195/A/. 17 Aug 1978. 27p. (In German).

The tube boiler heat exchanger has been designed for use in a steam generator circuit of a sodium cooled fast breeder reactor. U-type heat exchanger tubes with legs of different lengths are suspended within the boiler. They are made of steel with 9% chrome content. Each tube is passed through an opening in the tube wall with some clearance and then tightly sealed to the tube wall by means of a sleeve. This seal is achieved by a brazed connection using Microbraz 135 brazing metal (nickel base alloy) applied by means of a high frequency induction technique.

3474 NUCLEAR REACTOR. Howard, R.S.; Hodgson, D.O.; Dale, J.G.; Thomas, C. (to Nuclear Power Co. Ltd., London (UK); Deutsches Patentamt, Muenchen (Germany, F.R.)). German(FRG) Patent 2,805,188/A/. 17 Aug 1978. 18p. (In German).

In a sodium cooled fast breeder reactor the fuel assembly proper is separated by a partition wall. The inner wall is equipped with a thermal insulation. It consists of a large number of spaced layers of stainless steel plates. Each layer has square panels fixed on the wall surface in vertical and horizontal rows. The gaps between the panels are closed by cruciform covers. The arms of the covers overlap the opposite areas of adjacent panels. The cruciform cover strips, the panels and the spacers between the cover strips constitute labyrinth seals for the coolant.

3475 NUCLEAR REACTOR CONTROL. Ingham, R.V. (to UKAEA London Office). British Patent 1,521,615/A/. 16 Aug 1978. 5p.

The reactor control described is applicable to liquid metal cooled fast breeder nuclear reactors and provides that in the event of an emergency condition wherein the rate of coolant flow is reduced a rapid power set-back of appropriate magnitude is achieved by causing limited injection of control rods into the reactor, the insertion being sufficient to reduce reactivity to a safe level, and, subsequently, fine control at the reduced power output is achieved by automatic control.

3476 REACTOR. Toyama, M.; Kasai, S. (to Toshiba Corp., Kawasaki, Kanagawa (Japan)). Japanese Patent 53-92,095/A/. 12 Aug 1978. 4p. (In Japanese).

Purpose: To provide a LMFB type reactor

wherein effusion of coolants through a loop contact portion is reduced even when fuel assemblies float up, and misloading of reactor core constituting elements is prevented thereby improving the reactor safety. Constitution: The reactor core constituents are secured in the reactor by utilizing the differential pressure between the high-pressure cooling chamber and low-pressure cooling chamber. A resistance port is formed at the upper part of a connecting pipe, and which is connect the low-pressure cooling chamber and the lower surface of the reactor core constituent. This resistance part is formed such that the internal sectional area of the connecting pipe is made larger stepwise toward the upper part, and the cylinder is formed larger so that it profiles the inner surface of the connecting pipe.

- 3477 DEVICE OR INSTALLATION FOR USE IN A LIQUID ALKALI METAL ENVIRONMENT. Campbell, C.S. (to UKAEA London Office; Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,802,577/A/. 10 Aug 1978. 11p. (In German).

The invention relates to the use of materials in liquid alkali metals, especially sodium, for support faces. It is proposed to use as a complementary support face for stainless steel bearings a nickel base aluminized alloy. The invention is explained by the example of a heat exchanger for a liquid sodium cooled fast breeder reactor. In the design, fuel rods with stainless steel claddings are supported by cell type grids. In the invention, these grids are made of nickel base alloys and carry bushings with aluminized bores constituting the complementary support faces for the nuclear fuel rods.

- 3478 LIQUID METAL COOLED FAST BREEDER REACTOR. Durston, J.G.; Hind, J.R. (to Nuclear Power Co. Ltd., London (UK); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,804,560/A/. 10 Aug 1978. 11p. (In German).

The proposal relates to a change in the design of the inner vessel surrounding the fuel subassembly supported on a grid plate in a liquid metal cooled fast breeder reactor. In the invention, this vessel which is surrounded by liquid metal, carries a number of vertical ribs on its inner surface which are to serve the purpose of reducing the stresses acting on the wall of the vessel as a result of the differential temperatures in the coolant. A cover plate and baffles and guide vanes add to the effect of the ribs.

- 3479 TECHNICAL CONCEPT OF THE BREEDER NUCLEAR POWER STATION AND ITS REALIZATION IN THE FRG. Hueper, P. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). Technica (Basel); 27: No. 15, 1230-1231 (4 Aug 1978). (In German).

An introductory outline for the nonspecialist is presented.

- 3480 FAST SHUTDOWN DEVICE FOR A NUCLEAR REACTOR. Minnick, L.E. (to Electric Power Research Inst., Palo Alto, CA (USA); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,801,005/A/. 3 Aug 1978. 27p. (In German).

The fast shutdown effect is based on an intervention into the operative connection between control rod and control rod drive mechanism. During operation of the LMFBR the coolant sodium is developing sufficient differential pressure on the separating plate

and on the seal of the control rod to press the rod against the separating plate and to keep it above the neutron high-flux region of the core. If the flow rate of the coolant is reduced (accident, leak or rupture in the primary loop, earthquake) or the control rod is moved laterally the differential pressure on the separating plate decreases. The control rod drops because of gravity. A scram will follow.

- 3481 PRESSURE SENSOR FOR HIGH-TEMPERATURE LIQUIDS. Forster, G.A. (to Dept. of Energy). US Patent 4,103,555. 1 Aug 1978. Filed date 6 Jun 1977. 6p.

PAT-APPL-804,123.

A pressure sensor for use in measuring pressures in liquid at high temperatures, especially such as liquid sodium or liquid potassium, comprises a soft diaphragm in contact with the liquid. The soft diaphragm is coupled mechanically to a stiff diaphragm. Pressure is measured by measuring the displacement of both diaphragms, typically by measuring the capacitance between the stiff diaphragm and a fixed plate when the stiff diaphragm is deflected in response to the measured pressure through mechanical coupling from the soft diaphragm. Absolute calibration is achieved by admitting gas under pressure to the region between diaphragms and to the region between the stiff diaphragm and the fixed plate, breaking the coupling between the soft and stiff diaphragms. The apparatus can be calibrated rapidly and absolutely.

- 3482 FAST BREEDER REACTOR. ENERGY WITHOUT DEPLETION OF NATURAL RESOURCES. Vaughan, R.D. (Nuclear Power Co. (Risley) Ltd. (UK)); Farmer, A.A. (UK Atomic Energy Authority). Electr. Eng. (Johannesburg); 15-16, 18 (Aug 1978).

Primary energy growth during the next decades is predicted, and the role of the fast breeder reactor in relation to such a prediction is discussed. Attention is given to the fact that breeder reactors can provide energy without depleting natural uranium resources.

- 3483 (ANL-Trans--1140) DISTAV CALCULATION CODE FOR DETERMINING SODIUM TEMPERATURES IN DEFORMED HELICOID WIRE PIN BUNDLES IN FAST NEUTRON REACTORS. Leteturier, D.; Dufour, P.; Eyriey, M. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France); Societe ORDINA, 13 - Marseille (France)). Aug 1978. Translation of Paper NR-7 (CONF-780805--7). 18p. Dep. NTIS, PC A02/MF A01.

From 6. heat transfer conference; Toronto, Canada (7 Aug 1978).

The DISTAV (distribution of temperatures in an assembly during its lifetime) code is described. The code calculates sodium and pin sheath temperatures in a fast neutron reactor assembly which has been subject to deformations under irradiation is presented. The preliminary results obtained with this code are discussed and the experimental program required for its validation is described.

- 3484 OXYGEN GETTERS FOR FAST REACTOR FUEL PINS. Johnson, C.E.; Fee, D.C. (Argonne National Lab., IL). Nucl. Technol.; 40: No. 1, 89-97 (Aug 1978).

Oxidative attack of stainless-steel cladding in mixed uranium-plutonium oxide fast reactors may limit the life of the fuel pins. Present knowledge of cladding reactions suggests that the key factor in controlling cladding attack may be control of the oxygen potential (Δ anti

$G/\text{sub } O_2/ = RT \ln p/\text{sub } O_2/)$ by the insertion of oxygen getters within the fuel pin. Niobium, titanium, and vanadium are prime candidates as oxygen getters because they are easily oxidized at more negative oxygen potentials than the stainless-steel cladding. On the basis of predicted oxidation rates at the low oxygen pressures encountered in fuel pins, the recommended order of getter suitability is $V > Nb > Ti$.

- 3485 FUEL AND FISSION GAS RESPONSE TO SIMULATED THERMAL TRANSIENTS: EXPERIMENTAL RESULTS AND CORRELATION WITH FISSION GAS RELEASE AND SWELLING MODEL. Bandyopadhyay, G. (Argonne National Lab., IL). Nucl. Technol.; 40: No. 1, 62-78(Aug 1978).

To evaluate the role of fission gas in hypothetical core disruptive accidents, experimental and analytical information describing the fission gas behavior in rapid temperature transients is urgently needed. A direct-electrical-heating apparatus was used to obtain information on the fission gas behavior and the response of mixed-oxide fuel elements to simulated thermal transient conditions. The experimental results indicate that fission gas response and swelling behavior are strongly dependent on the transient heating rate, and that fission gas can contribute significantly to the failure of a fuel stack during a temperature transient. The microstructural results from these tests were subsequently used to perform a limited verification of the fission gas release and swelling code, FRAS, which was developed to describe the fission gas release behavior in rapid temperature transients. A comparison of the measured intragranular bubble sizes (and in some cases bubble densities) with the calculated bubble sizes (and densities) revealed that the current version of the FRAS code is inadequate in some transient conditions. A nonequilibrium of bubble coalescence may be necessary to describe the fission gas behavior in such temperature transients.

- 3486 EFFECTS OF RECENT MODELING DEVELOPMENTS IN PROMPT BURST HYPOTHETICAL CORE DISRUPTIVE ACCIDENT CALCULATIONS. Sienicki, J.J.; Abramson, P.B. (Argonne National Lab., IL). Nucl. Technol.; 40: No. 1, 106-115(Aug 1978).

The main objective of the development of multifield, multicomponent thermohydrodynamic computer codes is the detailed study of hypothetical core disruptive accidents (HCDAs) in liquid-metal fast breeder reactors. The main contributions such codes are expected to make are the inclusion of detailed modeling of the relative motion of liquid and vapor (slip), the inclusion of modeling of nonequilibrium/nonsaturation thermodynamics, and the use of more detailed neutronics methods. Scoping studies of the importance of including these phenomena performed with the parametric two-field, two-component coupled neutronic/thermodynamic/hydrodynamic code FX2-TWQPOOL indicate for the prompt burst portion of an HCDA that: (1) Vapor-liquid slip plays a relatively insignificant role in establishing energetics, implying that analyses that do not model vapor-liquid slip may be adequate. Furthermore, if conditions of saturation are assumed to be maintained, calculations that do not permit vapor-liquid slip appear to be conservative. (2) The modeling of conduction-limited fuel vaporization and condensation causes the energetics to be highly sensitive to variations in the droplet size (i.e., in the parametric values) for the sizes of interest in HCDA analysis. Care must therefore be exercised in the inclusion of this phenomenon in

energetics calculations. (3) Insignificant differences are observed between the use of space-time kinetics (quasi-static diffusion theory) and point kinetics, indicating again that point kinetics is normally adequate for analysis of the prompt burst portion of an HCDA. (4) No significant differences were found to result from assuming that delayed neutron precursors remain stationary where they are created rather than assuming that they move together with fuel. (5) There is no need for implicit coupling between the neutronics and the hydrodynamics/thermodynamics routines, even outside the prompt burst portion.

- 3487 BENCHMARK CALCULATIONS FOR A SODIUM-COOLED BREEDER REACTOR BY TWO- AND THREE-DIMENSIONAL DIFFUSION METHODS. II. Buckel, G.; Kuefner, K.; Stehle, B. (Kernforschungszentrum Karlsruhe, Ger.). Nucl. Sci. Eng.; 67: No. 2, 257-259(Aug 1978).

Results for our liquid-metal fast breeder reactor benchmark problem obtained by several diffusion computer programs for two- and three-dimensional reactor models showed unexplained behavior or discrepancies. With the help of additional calculations and comparisons, it was possible to improve some of the computer programs. Most of the original calculations have been repeated (and, in addition, some new calculations have been performed), and the revised results of the comparison are presented.

- 3488 FAST BREEDER REACTOR: A MILLENIUM OF PEACEFUL POWER. Compressed Air; 83: No. 8, 8-13(Aug 1978).

Proliferation concerns affecting the development of the LMFBR in the U.S. are reviewed and the search for more proliferation-resistant fuel cycles is described. Topics discussed include civilian reprocessing, breeder deferral, enrichment capacity, uranium supply, breeder economics, world energy needs, the Civex process, and institutional controls.

- 3489 QUANTITATIVE COMPARISON OF SOLIDIFICATION ANALYSIS FOR SINGLE-COMPONENT AND MIXED-OXIDE FUELS. Welch, T.D. (Oak Ridge National Lab., TN); Cronenberg, A.W. (Univ. of New Mexico, Albuquerque). Nucl. Sci. Eng.; 67: No. 2, 263-269(Aug 1978).

An important question to the liquid-metal fast breeder reactor safety program is a description of molten fuel dynamics, or, more specifically, whether fuel will freeze locally on structural material within the reactor core, preventing dispersal and nuclear shutdown, or in the extremities of the fuel assembly. A comparison is made between the solidification processes for single-component (i.e., UO_2) and mixed-oxide fuel [i.e., $(U,Pu)O_2$] by solving a Stefan-type problem for both pure and binary alloy solidification. Analytic calculations indicate that the freezing rate of the mixed fuel is not significantly different from that for the single-component system; thus, single-front analysis may be used for such mixed-oxide fuels in assessing safety questions associated with solidification phenomena.

- 3490 INVESTIGATION OF IRRADIATED URANIUM--PLUTONIUM CARBONITRIDE FUEL BY MICROPROBE ANALYSIS. Walker, C.T. (Joint Research Centre, Karlsruhe, Ger.). Nucl. Technol.; 39: No. 3, 289-296(Aug 1978).

A mixed carbonitride fuel irradiated to 3.9% fissions of initial metal atoms in a fast flux was examined by electron microprobe analysis. The fuel contained a large porous zone.

Inclusions at the edge of the zone contained palladium and tin and were formed from the liquid phase. Within the porous region, the platinum metals occurred in the UPd_3 and $U(Tc_{0.02}Ru_{0.75}Rh_{0.2}Pd_{0.06})_3C$ phases, molybdenum and technetium formed inclusions of the $(U,Pu)_{sub x}/(Mo,Tc)_{sub y}/C_{sub z}$ type, and part of the rare earth elements reacted with impurity oxygen to give a sesquioxide phase. Plutonium depletion was found near the fuel center, and plutonium enrichments were measured at the outer part of the porous zone and at healed cracks in the dense fuel. Cladding carburization occurred and, at the inner cladding surface, the carbon concentration was 0.4 wt %.

project have chosen to concentrate their assault on the plant design, stating that it is obsolete, that its size was improperly selected, and that it contributes to the risk of nuclear weapons proliferation to a greater extent than other moderately enriched fuel cycle concepts, such as the Shippingport light water breeder reactor. Arguments are presented that the design is up to date, its size was appropriately selected, and that contrary to contributing to nuclear weapons proliferation, it provides the United States an opportunity to implement the results of the International Fuel Cycle Evaluation (INFCE) in the near term, thereby assuring a leadership role in the worldwide nuclear community.

- 3491 RECENT IMPROVEMENTS IN IDENTIFYING FISSION PRODUCT SOURCES IN THE EXPERIMENTAL BREEDER REACTOR II. Lambert, J.D.B. (Argonne National Lab., IL); So, B.Y.C.; Kirn, F.S.; Armstrong, J.R.; Ebersole, E.R.; Laug, M.T. Nucl. Technol.; 39: No. 3, 275-282(Aug 1978).

The detection of a xenon tag released from an element with breached cladding has proved to be the best method of identifying a source of fission product release in the Experimental Breeder Reactor II. From experience gained from December 1974 to September 1977 in identifying the sources of 25 major releases, allowance can now be made for the composition changes in xenon tags brought about by irradiation and for the presence of more than one tag in the cover gas. This experience has also shown that calculated ratios of fission gas to tag gas in subassemblies can be used in identifying a source of release. An ancillary method, which uses the measured $^{131}Xe/^{134}Xe$ ratio in the cover gas, can also determine whether or not the source is uranium. The success of these recent improvements is reflected in the short time to identify a source and in the small number of other sources, frequently none, that are suspect.

- 3492 PRIMARY PIPE RUPTURE ACCIDENT ANALYSIS FOR THE CLINCH RIVER BREEDER REACTOR. Albright, D.C.; Bari, R.A. (Brookhaven National Lab., Upton, NY). Nucl. Technol.; 39: No. 3, 225-257(Aug 1978).

Analyses of primary pipe rupture accidents in the Clinch River Breeder Reactor were carried out with Brookhaven National Laboratory-modified versions of the DEMO code. The thermal transient responses of the core and radial blanket were calculated for a large number of initial conditions and plant configurations. These include studies of variations of pipe break size and operating power. Pipe ruptures commencing from two-loop initial operating conditions are presented. The sensitivity of the thermal response to variations in particular parameters within the estimated ranges of their uncertainties has been studied. Conditions under which sodium boiling in the core occurs are delineated.

- 3493 EVALUATION OF THE CLINCH RIVER BREEDER REACTOR PLANT DESIGN. Hill, E.H. (Proj Manage Corp). Power Eng.; 82: No. 8, 93-98(Aug 1978).

Recent months have seen the Clinch River Breeder Reactor Plant (CRBRP) Project become perhaps the most controversial and debated power issue in the United States. The project is faced with the prospect of termination, or at best a significant delay, either of which promises to place the United States well behind several other industrialized nations in breeder reactor development. The opponents of the

- 3494 THREE-DIMENSIONAL DEFLECTION ANALYSES OF WIRE-SPACED FUEL PIN BUNDLES UNDER TEMPERATURE AND HYDRODYNAMIC FORCE FIELDS WITH IRRADIATION EFFECTS. Sakai, K.; Okubo, Y.; Hishida, H. (Mitsubishi Atomic Power Industries, Inc., Omiya, Saitama (Japan). Engineering and Development Div.). Nucl. Eng. Des.; 48: No. 2, 595-610(Aug 1978).

An analytical method of evaluating the effects of non-uniform thermal expansion, hydrodynamic force acting on the periphery of a fuel pin and thermal and irradiation-induced creep and swelling on the three-dimensional deflection modes of a fuel pin bundle, as well as the deviation in engineering hot channel factor, are presented. The analysis consists mainly of deriving an expression for a flexibility matrix in terms of the general solution of a beam deflection equation with an arbitrary number of loads under either fuel pin to fuel pin or fuel pin to wrapper tube contact condition. The resulting matrix equation is solved for loads corresponding to all contact points, in terms of which the deflection modes are given. The drag coefficient for a wire-wrapped fuel pin in flowing fluid was investigated experimentally. Several cases of sample calculation show that for a prototype LMFBR fuel subassembly consisting of 169 fuel pins, the engineering hot channel factor accounting for the three-dimensional fuel pin bundle deflection is around 1.023-1.035 at zero irradiation and further increase to 1.045 after 500 day irradiation. The maximum load due to pin contact and the order of pin bundle deflection increase according to the irradiation level.

- 3495 PROBABILISTIC DISTRIBUTIONS OF PIN GAPS WITHIN A WIRE-SPACED FUEL SUBASSEMBLY AND SENSITIVITIES OF THE RELATED UNCERTAINTIES TO PIN GAP. Sakai, K.; Hishida, H. (Mitsubishi Atomic Power Industries, Inc., Omiya, Saitama (Japan). Engineering and Development Div.); Nakai, Y.; Saito, S. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Nucl. Eng. Des.; 48: No. 2, 505-515(Aug 1978).

Probabilistic fuel pin gap distributions within a wire-spaced fuel subassembly and sensitivities of the related uncertainties to fuel pin gaps are discussed. The analyses consist mainly of expressing a local fuel pin gap in terms of sensitivity functions of the related uncertainties and calculating the corresponding probabilistic distribution through taking all the possible combinations of the distribution of uncertainties. The results of illustrative calculations show that with the reliability level of 0.9987, the maximum deviation of the pin gap at the cladding hot spot of a center fuel subassembly is 8.05% from its nominal value and the corresponding probabilistic pin gap distribution is shifted to the narrower side due to the external confinement of a pin bundle with a wrapper

tube.

3496 ANALYSIS OF RADIOACTIVE CORROSION PRODUCT TRANSFER IN SODIUM LOOP SYSTEMS.

Polley, M.V.; Skyrme, G. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). J. Nucl. Mater.; 75: No. 2, 226-237(Aug 1978).

A solution-precipitation model for radioactive corrosion product transfer in LMFBR primary circuits and experimental sodium loops is discussed. Expressions are given for the quantity released to flowing sodium from steel undergoing bulk corrosion. The observed super-stoichiometric release of ^{54}Mn and sub-stoichiometric release of cobalt isotopes are attributed to high and low relative solubilities in sodium, respectively. It is shown that relative solubility values are required for calculating ^{58}Co and ^{60}Co release but not for calculating ^{54}Mn release. An expression is given for the quantity deposited onto a surface undergoing either bulk corrosion or bulk deposition. This is reduced to a table of simple relationships which is used to interpret experimental data. It is thereby concluded that ^{60}Co deposition is controlled by diffusion in the liquid and ^{54}Mn deposition is likely to be controlled by solid-state diffusion at high temperatures and bulk deposition at low temperatures.

3497 FAST BREEDER REACTORS, CURRENT STATUS AND DEVELOPMENT TRENDS.

Kessler, G. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueiter). Kerntechnik; 20: No. 8, 349-356(Aug 1978). (In German).

The prototype fast breeder reactors Phenix, PFR and BN-350 have so far shown satisfactory operating performance and have reached a maximum fuel burnup of 75,000 Mwd/t. FFTF, BN-600 and SNR-300 are still under construction and are scheduled for commissioning in 1980-83. In France, the construction of the power plant Super-Phenix was started in 1977. In Britain and the USSR large breeder reactor power plants of 1300 and 1600 MW respectively are still in the detailed planning stage. The research and development programmes pursued concurrently with this construction activity are concentrated on reactor physics and safety engineering investigations of large cores and on the development of steels with low swelling and creep at high neutron fluences. (orig.) 891 HP.

3498 ANALYSIS OF THE EFFECT OF EMERGENCY OPERATING REGIMES OF A FAST POWER REACTOR ON THE FUEL PERFORMANCE.

Proshkin, A.A.; Bagdasarov, Yu.E.; Zabud'ko, L.M.; Buksha, Yu.K.; Likhachev, Yu.I. At. Energ. (USSR); 45: No. 2, 98-104(Aug 1978). (In Russian).

Performance of the BN-600 reactor type fuel elements have been analysed in different emergency regimes including power supply failure, cooling circuit cut-off and reactivity insertions. The stress-strain states have been calculated in terms of two models assuming strong fuel-to-cladding bonding (1) and mere contact (2). Fuel element performance criteria are formulated. According to the calculations the contribution of the emergencies to the fuel element failure rate in the model 2 is higher than in the model 1. In this case, the reactor accidents with faultless operation of corresponding emergency safety systems contribute little to the rate, should the accidents occur up to 5 times during the reactor life time.

3499 FAST BREEDER REACTOR.

Brandstetter, A. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.); Internationale Natrium-Brutreaktor-Bau G.m.b.H., Bergisch Gladbach (Germany, F.R.)). Tech. Mitt.; 71: No. 8, 439-446(Aug-Sep 1978). (In German).

Sodium as a coolant, technology of sodium, cooled fast breeders, safety behaviour, the plutonium problem, and their role in energy policy are discussed. (orig.) 891 RW.

3500 FAST BREEDER REACTOR.

Brandstetter, A. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.); Internationale Natrium-Brutreaktor-Bau G.m.b.H., Bergisch Gladbach (Germany, F.R.)). Tech. Mitt.; 71: No. 8, 439-446(Aug-Sep 1978). (In German).

Sodium as a coolant, technology of sodium, cooled fast breeders, safety behaviour, the plutonium problem, and their role in energy policy are discussed.

3501 CONSTRUCTION OF SUPER-PHENIX AND THE EUROPEAN COOPERATION IN THE FIELD OF BREEDERS.

Ertaud, A.; Naigeon, J. (Novatome Industries, 92 - Le Plessis-Robinson (France)). Rev. Gen. Nucl.; No. 4, 327-332(Aug-Sep 1978). (In French).

Laying stress on the essential part which can be played by the breeders to meet with power requirements, the authors describe the various steps of european cooperation in this field, and the industrial and commercial organization which has been setup for the development of Super-Phenix.

3502 COMPUTER CONTROL SYSTEM FOR SODIUM TEST LOOPS.

Takamatsu, S.; Okano, H.; Shigemasa, T. (Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan). Research and Development Center); Satoh, M.; Satoh, T. J. Nucl. Sci. Technol. (Tokyo); 15: No. 8, 612-620(Aug 1978).

A digital computer system was developed for automatic operation of the Toshiba Sodium Test Loops, intended for studying the control system of liquid metal fast breeder reactors. The test installation is controlled by a system of two TOSBAC-40C central processing units, which take care of plant monitoring, data logging, information display, sequence control and closed loop control. The operator console linked to the computers can handle more than 3,000 process input/output signals. The state of the plant and measured data are displayed on two color cathode-ray tube installed on the console. Normal operation of the four loops constituting the installation can be supervised by one single operator. Safety in the case of power failure and other emergencies is assured by a safety protection system that functions independently of the computer system.

3503 APPLICATION OF GALERKIN METHOD TO HYDRAULIC ANALYSIS FOR TURBULENT FLUID FLOW AROUND LMFBR FUEL ROD.

Shimizu, T.; Kawamura, H. (Nippon Atomic Industry Group Co. Ltd., Kawasaki, Kanagawa. Nuclear Research Lab.). J. Nucl. Sci. Technol. (Tokyo); 15: No. 8, 628-631(Aug 1978).

ANISOTROPY; COOLANTS; FLOW MODELS; FUEL RODS; GALERKIN-PETROV METHOD; HYDRAULICS; LMFBR TYPE REACTORS; TURBULENT FLOW; TWO-DIMENSIONAL CALCULATIONS; VELOCITY

3504 REPROCESSING FAST REACTOR FUEL IN

BRITAIN. Nucl. Eng. Int.; 23: No. 275, 45-46(Aug 1978).

The existing reprocessing plant at Dounreay is being modified and extended to accommodate fuel from PFR in 1979. The changes are necessary because of higher rating, burn-up, and plutonium content. Sodium must also be removed from the fuel-assemblies before reprocessing. Changes in flow sheet, instrumentation, design of certain components, use of lasers for dismantling, and waste management are discussed briefly. Reference is made to R and D work in progress.

3505 PIPES IN STEEL OR SPECIAL MATERIALS FOR NUCLEAR BOILERS. Delcroix, J. (Vallourec, 75 - Paris (France)). Rev. Gen. Nucl.; No. 4, 277-281(Aug-Sep 1978). (In French).

Some of the main achievements among the tubular products used for the construction of nuclear power stations are described i.e.: housing tubes, steam generator tubes, welded tubes for heat exchangers and condensers. The quality control organization which has been set up for the development of this equipment is presented.

3506 CHARACTERIZATION OF LMFBR FUEL-SODIUM AEROSOLS. Allen, M.D.; Briant, J.K. (Battelle Pacific Northwest Labs., Richland, WA (USA)). Health Phys.; 35: No. 2, 237-254(Aug 1978).

Liquid Metal Fast Breeder Reactor (LMFBR) fuel-sodium condensation aerosols which may be formed in a major reactor accident have been investigated. A pulsed beam from a high-powered CO₂ laser vaporized PuO₂-UO₂ from the surface of a rotating LMFBR fuel pellet. The PuO₂-UO₂ vapor rapidly condensed and coagulated into branched, chain-like aggregates. These aggregates were swept by argon into a sodium vapor atmosphere. The sodium condensed on the aggregates and the mixed aerosol was introduced into 50% relative humidity air. These aerosols have been characterized according to physical appearance, crystalline structure, and aerodynamic behavior. The mixed PuO₂-UO₂-Na aerosol appeared physically similar to sodium aerosol alone. X-ray and electron diffraction analyses indicated that no chemical interaction occurred between the PuO₂-UO₂ aggregates and sodium aerosol, or at least that the product of a reaction is amorphous, if they were kept in a fairly dry atmosphere; however, when exposed to water vapor, sodium uranyl tricarbonate was formed. The activity median aerodynamic diameter (AMAD) of the PuO₂-UO₂ aggregates was approximately 0.80µm, depending somewhat on the energy input per laser pulse. The AMAD of the mixed PuO₂-UO₂-Na aerosol was higher than for the PuO₂-UO₂ aerosol alone, and increased as the sodium-to-mixed oxide ratio increased.

3507 MEASUREMENT OF TRITIUM AND TRITIATED WATER VAPOUR IN NUCLEAR POWER INSTALLATIONS. I. TRITIUM FORMATION AND TRANSPORT IN NUCLEAR POWER INSTALLATIONS. Gilar, O.; Kubalek, J.; Sedakov, V. (Tesla, Premysleni (Czechoslovakia)). Vyzkumny Ustav Pristroju Jaderne Techniky. Jad. Energ.; 24: No. 8, 290-296(Aug 1978). (In Russian).

The formation of tritium is described in heavy water-cooled, WWER-type and LMFBR-type reactors. In heavy water reactors tritium results, among others, as the product of reaction D(n,γ)T and its amount depends on the reactor output and the duration of D₂O irradiation. In a reactor with a mean neutron flux density of 10¹³ cm⁻²s⁻¹ after one year of operation in 1 kg D₂O about 1.85x10¹⁰ s⁻¹ tritium is formed, this in the form of gas or oxides. In WWER type reactors tritium results

from uranium fission in fuel elements and neutron capture by boron. It has the form of HTO or tritiated water vapours. In a power unit fitted with two WWER reactors with an output of 2x500 MWsub(th), the amount of tritium in the primary circuit is stabilized at 3.7x10¹⁰ s⁻¹m⁻³ and in the reactor water regime at 7.4x10⁹ s⁻¹m⁻³. In LMFBR reactors tritium is formed also from actinide fission and by neutron capture in boron and lithium. Tritium activity is in fast reactors higher than in PWR's and its transport depends on the content of tritium in the primary circuit and in the coolant cover atmosphere. Tritium formation is also described in research reactors, reprocessing facilities, in radiochemical and radiological laboratories, and in particle accelerator laboratories.

3508 OUTLINE OF THE PRELIMINARY WORKING DESIGN (1) OF PROTOTYPE FBR 'MONJU'. Doryokuro Giho; No. 27, 78-104(Aug 1978). (In Japanese).

Following the conceptual design, the preliminary working design (1) of the prototype fast breeder reactor 'Monju' has been completed by five industrial firms. The design connects the conceptual design to the actual working design. In this design, therefore, the designs were detailed in coordination of machinery, pipings, wirings, etc. with building structures. Prior to the design, recheck was made on the basis of the results of design studies and research and development, and the specifications were partially modified. It serves also for the purpose of governmental safeguard inspection. The features of the prototype FBR Monju are heat output of 714 MW, electric output of 300 MW, fuel of PuO₂-UO₂, breeding ratio of 1.2, and coolant of sodium. The following matters are described: reactor core, reactor construction, fuel handling system, heat transfer system, steam generating system, turbo-generator system, plant operation mode, electric, instrumentation and control equipments, radioactive waste treatment facility, layout, buildings and containment, etc..

3509 NEW WELDING METHOD FOR TUBES AND TUBE PLATES OF HEAT EXCHANGERS FOR FBRs. Shibato, E.; Koyama, T. (Hitachi Ltd., Ibaraki (Japan)). Hitachi Works. Yosetsu Gijutsu; 26: No. 8, 72-76(Aug 1978). (In Japanese).

The cooling systems for FBRs comprise the intermediate heat exchanger for exchanging heat between primary sodium and secondary sodium and the steam generator for exchanging heat between the secondary sodium and water. The intermediate heat exchanger is that of shell and tube type, and it becomes large as the reactor proceeds to a demonstration reactor and a commercial reactor. Accordingly, the number of heat transfer tubes and the size of tube plates increase, and the welding method for tubes and tube plates to obtain the high reliability of welded parts and to achieve low cost and short delivery is required. For this purpose, the method of inserting heat transfer tubes into tube plates by same dimension as their wall thickness and welding them by automatic TIG method so as to attain perfect penetration was developed. In order to confirm the reliability of the welded joints by this method, the test for evaluating the short time and long term strength was carried out. As the result, it was confirmed that the satisfactory performance of the welded joints was able to be obtained, and this method can be applied to actual heat exchangers. Besides, non-destructive inspection technique required for the development of this welding method was

developed. The determination of welding torch position and the improvement of welding efficiency are being studied.

- 3510 (DOE-tr--27) FUEL QUANTITIES, ACTIVITY INVENTORIES, AND RADIOACTIVE RELEASES FROM LIGHT-WATER REACTORS AND SODIUM-COOLED FAST BREEDERS. Papp, R. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Abt. fuer Angewandte Systemanalyse). 28 Jul 1978. Translation of KFK--2453, June 1977 51p. Dep. NTIS (US Sales Only), PC A04/MF A01. Work performed under United States--W. Germany Fast Reactor Exchange Program.

A survey is given of mass flows and nuclide inventories in the fuel cycles in light-water and breeder reactors. Data are also given for typical quantities of radioactive waste. However, this report is mainly concerned with the radioactive harmful materials released to the environment when energy is produced on the basis of these reactor systems. In spite of the uncertainty associated with the data, it can be said that on the basis of the quantities which pertain to the environment in comparison to the thermal reactor, the fast breeder reactor does not represent a new category of environmental impact.

- 3511 POSITIONING AND LOCKING DEVICE FOR FUEL PIN TO GRID ATTACHMENT. Frick, T.M.; Wineman, A.L. (to Department of Energy, Washington, DC (USA)). Canadian Patent 1,035,474/A/. 25 Jul 1978. 10p.

The invention comprises an elongated fuel pin assembly having a fuel pin lower end cap, said end cap having a keyed slot perpendicular to its major axis; a positioning and locking bar adapted to slidably mate with a first part of said keyed slot, said bar having multiple, spaced apart, notches exposed along its length; a thin wall fuel pin attachment and flow distribution block disposed below said fuel pin assembly, said block having an upper edge adapted to support said positioning and locking bar at each extremity thereof and having spaced apart elongated keyed slots in the walls thereof, each of said slots having its major axis parallel to the major axis of said assembly; a fuel pin attachment grid beam adapted to slidably mate with said keyed slot of said fuel assembly whereby multiple fuel pin assemblies are originally maintained in locked spaces apart rows.

- 3512 METHOD OF SHIELDING A LIQUID-METAL-COOLED REACTOR. Sayre, R.K. (to Dept. of Energy). US Patent 4,102,739. 25 Jul 1978. Filed date 30 Sep 1976. 6p. PAT-APPL-728,359.

The primary heat transport system of a nuclear reactor - particularly for a liquid-metal-cooled fast-breeder reactor - is shielded and protected from leakage by establishing and maintaining a bed of a powdered oxide closely and completely surrounding all components thereof by passing a gas upwardly therethrough at such a rate as to slightly expand the bed to the extent that the components of the system are able to expand without damage and yet the particles of a the bed remain close enough so that the bed acts as a guard vessel for the system. Preferably the gas contains 1 to 10% oxygen and the gas is passed upwardly through the bed at such a rate that the lower portion of the bed is a fixed bed while the upper portion is a fluidized bed, the line of demarcation therebetween being high enough that the fixed bed portion of the bed serves as guard vessel for the system.

- 3513 FAST NEUTRON REACTOR. Berniolles, J.M.; Jogand, P. (to Commissariat a l'Energie Atomique). US Patent 4,101,377. 18 Jul 1978. Priority date 29 Mar 1976, France, 8p.

A cylindrical inner vessel containing the reactor core is placed coaxially within a main vessel. Heat exchangers are disposed within the annular space formed between the two vessels, each heat exchanger being delimited by an outer shell and a coaxial inner shell which surrounds a central feed duct having a vertical axis. The upper end of the annular space communicates with a flared end portion of the central duct through a first circular window formed in the inner shell. The lower end of the annular space communicates with the region located between the inner vessel and the main vessel through a second circular window formed in the outer shell.

- 3514 (DOE-tr--178) HEAT TRANSMISSION DURING START-UP IN CARBIDE FUEL RODS PROVIDED WITH AVERAGE TO LARGE FABRICATION GAPS. Steiner, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). 5 Jul 1978. Translation of KFK--2472, June 1977 35p. Dep. NTIS, PC A03/MF A01.

Work performed under United States--W. Germany Fast Reactor Exchange Program.

The effect of various geometries on the temperature distribution in helium bonded carbide fuel rods was investigated by solving the heat-conduction equation by means of a Fourier series development. In particular, concentric, eccentric, and staggered arrangements of uncracked pellets as well as pellet cracking with subsequent displacement of pellet fractions toward the cladding were described by means of a model. It turned out that the heat transmission coefficients, determined experimentally during start-up in helium-bonded carbide fuel rods with average fabrication gaps, can only be explained by this last effect. The fact that this effect has not been taken into consideration is also the reason for the discrepancy between the experimental results and the predictions by the GAPRS heat transmission model which is based on a concentrically placed, uncracked pellet.

- 3515 (DOE-tr--175) ANALYSIS OF THE STRUCTURE AND THE THERMAL BEHAVIOR OF THE UO_2 -- PuO_2 FUEL IN THE IRRADIATION EXPERIMENT FR2 CAPSULE TEST SERIES 5A. Lopez Jimenez, J.; Elbel, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). 3 Jul 1978. Translation of KFK--2477, June 1977 50p. Dep. NTIS, PC A03/MF A01.

Work performed under United States--W. Germany Fast Reactor Exchange Program.

In the Karlsruhe FR2 research reactor nine fuel pins were tested in a total of three irradiation capsules within the framework of the 5a test series. The fuel pins contained UO_2 -- PuO_2 fuel pellets. The irradiation times were different for the three capsules. They reached burnup values of approximately 6, 17, and 47 MWd/kgMe for rod power levels of 400 to 600 W/cm and cladding surface temperatures between 500 and 700°C. A detailed analysis of the fuel structure was carried out in order to determine the fuel surface temperature. For this purpose two models were used which describe the formation of the columnar crystal zone or the zone where the directed growth of the grain depends upon the irradiation time and the temperature profile in the fuel.

3516 (DOE-tr--174) FAST BREEDER CLADDING TUBES PROVIDED WITH HELICAL FINS: DEVELOPMENT AND TESTING. Jacobi, O. (comp.). (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). 1 Jul 1978. Translation of KFK--2442, April 1977 69p. Dep. NTIS, PC A04/MF A01.

Work performed under United States--W. Germany Fast Reactor Exchange Program.

Development of fast breeder tubes with helical fins started around 1966. The development was initiated mainly by the search for a more favorable alternative to the conventional spacers such as grids and spiral wires. First of all the possibility was investigated of fabricating them on an industrial scale. The first thermohydraulic and fluid dynamic tests were made in parallel by AEG, GfK, and Interatom. The different stages of development are shown and, finally, two rather large projects are described. These projects are the licensing for the irradiation of finned-tube oxide fuel elements in KNK-II and the already complete 19-rod fuel element bundle irradiation experiment Mol 7D GfK performed in the 500 kW sodium loop of the Belgian BR2 reactor. In this BR2 experiment the maximum target burnup of 85,000 MWD/t of oxide was attained without complications.

3517 LIQUID-METAL FAST BREEDER REACTOR FUEL ROD PERFORMANCE AND MODELING AT HIGH BURNUP. Verbeek, P. (Belgonucleaire, Brussels); Toebbe, H.; Hoppe, N.; Steinmetz, B. Nucl. Technol.; 39: No. 2, 167-185 (Jul 1978).

The fuel rod modeling codes IAMBUS and COMETHE were used in the analysis and interpretation of postirradiation examination results of mixed-oxide fuel pins. These codes were developed in the framework of the SNR-300 research and development (R and D) program at Interatom and Belgonucleaire, respectively. SNR-300 is a liquid-metal fast breeder reactor demonstration plant designed and presently constructed in consortial cooperation by Germany, Belgium, and the Netherlands. RAPSODIE I, the two-bundle irradiation experiment, was irradiated in the French test FBR RAPSODIE FORTISSIMO and is one of the key irradiation experiments within the SNR-300 R and D program. The comparison of code predictions with postirradiation examination results concentrates on clad diameter expansions, clad total axial elongations, fuel differential and total axial elongations, fuel restructuring, and fission gas release. Fuel rod modeling was considered in the light of benchmarking of the codes, and there was consideration of fuel rod design for operation at low and high burnup.

3518 EFFECT OF PRIOR IRRADIATION CREEP ON THE SUBSEQUENT BURST STRENGTH OF 20% COLD-WORKED TYPE 316 STAINLESS-STEEL CLADDING. Duncan, D.R. (Hanford Engineering Development Lab., Richland, WA). Nucl. Technol.; 39: No. 2, 199-206 (Jul 1978).

Creep rates of metals can be greatly enhanced by neutron irradiation experienced in a fast reactor environment. Because irradiation-induced creep strains can be large in magnitude for duct and cladding components, the effect of irradiation creep on subsequent mechanical property behavior must be quantified to provide a design base for core component performance assessment. Pressurized tubes that had sustained a peak value of irradiation creep strain of 1.04% and peak fast fluences of 10×10^{22} n/cm² under irradiation in the Experimental Breeder Reactor II from 650 to 857 K were tested in subsequent transient burst loading. Results of testing show that

subsequent deformation is unaffected by prior irradiation creep within experimental error.

3519 HYDROGEN BURDEN FROM THE STEAM SIDE CORROSION IN SODIUM-HEATED STEAM GENERATORS. Roy, P.; Rodgers, D.N. (General Electric Co., San Jose, CA). Nucl. Technol.; 39: No. 2, 213-215 (Jul 1978).

It has been observed that in sodium-heated steam generators most of the hydrogen produced from water- or steam-side corrosion permeates into the secondary sodium. The amount of hydrogen that diffuses into the secondary side is extremely important from the standpoint of the design of the cold trap, hydrogen background levels, and tritium concentrations in the secondary system. We have compiled all the available data on H₂ permeation from all operating sodium-heated steam generators. In addition, based on the available steam-side corrosion data, we have made tentative recommendations: Steady-state H₂ flux, (1) Evaporator (2 1/4 Cr--1 Mo)--1.8 to 2.2 $\times 10^{-11}$ g H/cm².s. (2) Superheater (2 1/4 Cr--1 Mo)--1.2 to 1.7 $\times 10^{-11}$ g H/cm².s. Initial H₂ flux (up to 2000 h): Evaporator and superheater (2 1/4 Cr--1 Mo)--approximately 7 to 8 $\times 10^{-11}$ g H/cm².s.

3520 ELECTROCHEMICAL OXYGEN SENSOR FOR MEASUREMENT OF OXYGEN IN LIQUID SODIUM. Roy, P.; Bugbee, B.E. (General Electric Co., San Jose, CA). Nucl. Technol.; 39: No. 2, 216-218 (Jul 1978).

The results obtained from approximately 4000 h of operation of an electrochemical oxygen meter developed at General Electric Company for measuring oxygen activities in liquid sodium are described. In the present design, a small ceramic cup (ThO₂ with 7 to 8 wt % Y₂O₃) has been brazed to a metal tube that is inserted into the liquid sodium via a standard sodium flange. The reference electrode used in the present design consists of In/In₂O₃. Several meters were tested in a flowing sodium system, where the oxygen concentrations were controlled by a cold trap. At 500°C, the oxygen electromotive force (emf) output can be expressed as $\text{emf}/\text{sub (V)} = 0.269 + (160/T) (+ - 0.003 \text{ V})$, where T is the cold-trap temperature in K. The temperature coefficient of the meters can be expressed as $(dE/dT) = 0.51 \pm (0.02) \text{ mV}/^\circ\text{C}$.

3521 MECHANISTIC STUDY OF FUEL FREEZING, CHANNEL PLUGGING, AND CONTINUED COOLABILITY DURING FAST REACTOR OVERPOWER EXCURSIONS. Wong, K.W.; Dhir, V.K.; Kastenber, W.E. (Univ. of California, Los Angeles). Nucl. Technol.; 39: No. 2, 121-137 (Jul 1978).

A mechanistic model that describes events leading to in-channel fuel plateout following fuel pin failure under slow transient overpower accident conditions has been proposed. The thermal and hydraulic effects of the plated out fuel have also been investigated. Results based on this model indicate that during a slow transient overpower accident (2.4 cent/s), in-core fuel freezing and channel plugging can be expected to occur. The extent of fuel freezing and channel plugging is sensitive to fuel particle size, the amount of fuel injected, and the amount of fission gas injected into the coolant channel. The results are also compared with experimental data from the TREAT H-4 experiment.

3522 (BNWL-tr--335) BEHAVIOR OF STANDARD RAPSODIE FORTISSIMO FUEL AT HIGH BURNUP. Blanchard, P.; Lambert, G.; Millet, P.; Mikailoff, H. (International Atomic Energy

Agency, Vienna (Austria)). Jul 1978.
Translation of IAEA-SM--225/8 (CONF-780407--
7). 21p. Dep. NTIS (US Sales Only), PC A02/
MF A01.

From Symposium on design, construction and
operating experience of demonstration liquid
metal fast breeder reactors; Bologna, Italy (10
Apr 1978).

Results are presented which were obtained by
destructive examination of a complete fuel
bundle of 61 rods with 10 at. % burnup from the
Rapsodie Fortissimo reactor. A set of
micrographic sections was cut through the
bundle at linear power levels of 290 to 400 W/
cm. The temperature distributions evaluated at
a given height, starting from the positions at
the limit of the zones containing columnar
grains, are remarkably similar from rod to rod
(mean deviation ca. 25°C), whereas the central-
hole diameters are strongly disperse. These
dispersions are clearly greater than expected
from the range of data on fuel fabrication. It
is shown that they are due to several factors:
random radial distribution of the clearance in
the fissures, variable swelling of the steel of
the cans and variable in-pile sintering of the
oxide. In-pile sintering of pellets and
swelling of cans lead to overheating of the
oxide surface and to continuous evolution of
the structures. However, the central
temperatures continue to be low, and below
their value at the start of the service life.

3523 (BNWL-tr--339) SEPARATION OF
RADIOACTIVE IMPURITIES BY THE COLD TRAP OF THE
BOR-60 REACTOR. Kizin, V.D.; Lisitsyn, E.S.;
Poznyak, G.I.; Polyakov, V.I.; Chechetkin,
Yu.V. Jul 1978. Translation of NIIAR-P-2,
No. 296, 1977 27p. Dep. NTIS, PC A03/MF A01.

The separation of radioactivity in the
regular cold trap of the BOR-60 reactor was
studied experimentally. It was shown that ^{65}Zn ,
 ^{134}Cs , ^{136}Cs , ^{137}Cs , ^{131}I , ^{60}Co and ^{124}Sb
accumulate in this trap. The sodium is purified
almost completely of iodine in the cold trap,
and the radioactivity of the cesium isotopes
there amounts to 20-30% of their content in the
primary loop. Experiments were carried out to
determine the quantitative characteristics of
the processes of separation of cesium from
sodium by sorption on the sodium oxides and
metal surfaces.

3524 (BNWL-tr--334) ANALYSIS OF THE
PERFORMANCES OF THE ASSEMBLIES AND OF THE CORE
OF PHENIX. Ravier, J.; Delpeyroux, P.;
Estavoyer, M.; Bernard, B.; Carnoy, M.;
Courcon, P.; Jallade, M.; Ollier, H.; Coulon,
P.; Dupouy, J.M. (International Atomic Energy
Agency, Vienna (Austria)). Jul 1978.
Translation of IAEA-SM--225/9 (CONF-780407--
9). 23p. Dep. NTIS, PC A02/MF A01.

From Symposium on design, construction and
operating experience of demonstration liquid
metal fast breeder reactors; Bologna, Italy (10
Apr 1978).

The recent operating characteristics of the
PHENIX Reactor are described and the core and
fuel assembly performance is discussed.

3525 KERNKRAFTWERK KALKAR (SNR-300) =
CONSTRUCTION OF THE INERTING SYSTEMS WJ01 AND
WJ02 AND THE TECHNICAL HEAVY-CURRENT EQUIPMENT
FOR THE POWER SUPPLY OF THE PROTECTION SYSTEM.
RECOMMENDATION OF THE
REAKTORSICHERHEITSKOMMISSION DURING ITS 134TH
MEETING ON MAY 17TH, 1978. Bundesanzeiger; 30:
No. 132, 1-2(Jul 1978). (In German).
CONSTRUCTION; COVER GAS; POWER SUPPLIES;
REACTOR PROTECTION SYSTEMS; RECOMMENDATIONS; SNR-
1 REACTOR

3526 (JAPFNR--316) FAST BREEDER REACTOR
DEVELOPMENT IN JAPAN. PROGRESS REPORT,
OCTOBER--DECEMBER 1977. (Power Reactor and
Nuclear Fuel Development Corp., Tokyo (Japan)).
Jul 1978. Translation of PNC-N--251-78-02
13p. Dep. NTIS (US Sales Only), PC A02/MF
A01.

Work performed under United States--Japanese
Fast Reactor Exchange Program.

Progress in LMFBR research and development
is reported in the following areas: JOYO
development, MONJU development, reactor
physics, structural components, instrumentation
and control, sodium technology, fuels and
materials, safety, and steam generators.

3527 INFLUENCE OF THE CHEMICAL COMPOSITION OF
AUSTENITIC STAINLESS STEELS ON THEIR CORROSION
BEHAVIOUR IN HIGH-TEMPERATURE SODIUM.
Eremias, B.; Fresl, M. (G.V. Akimov State Res
Inst of the Prot of Mater, Prague, Czech). J.
Nucl. Mater.; 75: No. 1, 186-192(Jul 1978).

Results of weight losses, carbon analysis
and mechanical properties after exposure to
sodium as well as argon atmosphere are given.
The five steels can be sorted into several
families, dependent on their composition. 15
refs.

3528 SWELLING OF TYPE-316 STAINLESS STEEL AT
HIGH FLUENCES IN EBR-2. Kenfield, T.A.;
Appleby, W.K.; Busboom, H.J.; Bell, W.L. (GE,
Sunnyvale, Calif). J. Nucl. Mater.; 75: No.
1, 85-97(Jul 1978).

Density determinations showed swellings of
up to 15% $\Delta V/V$ for 20% cold worked
steel at a neutron fluence level of $1.4 \times 10^{23} \text{ n/cm}^2$,
 $E > 0.1 \text{ MeV}$ (70 dpa). The peak swelling
temperature range was 550°C-600°C regardless of
the extent of cold working. Increasing the cold
work level reduced the swelling and tended to
broaden the swelling temperature peak. Cold
working reduced the average void sizes. 18
refs.

3529 SWELLING OF TYPE-316 STAINLESS STEEL AT
HIGH FLUENCES IN EBR-II. Kerfield, T.A.;
Appleby, W.K.; Busboom, H.J. (General Electric
Co., Sunnyvale, Calif. (USA). Fast Breeder
Reactor Dept.); Bell, W.L. (General Electric
Co., Pleasanton, Calif. (USA). Vallecitos
Nuclear Center). J. Nucl. Mater.; 75: No. 1,
85-97(Jul 1978).

A test to measure swelling induced by fast
neutron irradiation in unstressed specimens of
type-316 stainless steel has completed
irradiation in the EBR-II reactor. Results are
reported and discussed which describe the
swelling as a function of neutron fluence,
temperature of irradiation and extent of cold
work in the alloy. Density determinations
showed swellings of up to 15% $\Delta V/V$ for
20% cold worked type-316 stainless steel at a
neutron fluence level of $1.4 \times 10^{23} \text{ n/cm}^2$, $E > 0.1$
 MeV (70 dpa). The peak swelling temperature
range was 550°C-600°C regardless of the extent
of cold working. Increasing the cold work level
reduced the swelling and tended to broaden the
swelling temperature peak. Transmission
electron microscopy (TEM) investigations showed
that cold working had reduced the average void
sizes compared to those observed in the
solution annealed material.

3530 BREEDERS: OPERATIONAL EXPERIENCE WITH
FAST POWER REACTORS IN FIVE STATES - MORE
INTENSIVE GERMAN-FRENCH BREEDER COOPERATION.
Hueper, R. (Kernforschungszentrum Karlsruhe

G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). At. Strom; 24: No. 4, 103-109 (Jul-Aug 1978). (In German).

In the past year contracts have been signed and implemented for German-French cooperation in LMFBR development and commercialization. The first German nuclear power station with a sodium cooled fast reactor, KNK II in Karlsruhe, is going into operation. Construction of the prototype SNR 300 at Kalkar (Lower Rhine) is slowing down awaiting a decision of the German Federal Constitutional Court. On the international level, remarkable experience in the operation of fast power reactors has accumulated. Possible fuel cycle alternatives are being evaluated by an international committee.

3531 STOP ON CONSTRUCTION OF FAST BREEDER AT KALKAR REFUSED. Energiewirtschaft. Tagesfragen; 28: No. 7, 450-453 (Jul 1978). (In German).

The application of a plaintiff was refused by the verdict of the High Court for North Rhine-Westphalia of 16.6.1978, which would have restored the delaying effect of his application against the defendant Minister of the Land (his opponent) on the first part licence of the Kalkar nuclear power station. The plaintiff had to bear the costs of the proceedings, including extra legal costs of the co-defendant nuclear power company. The value of the object of the case was set at DM 10,000. This means that construction can continue. The most important reasons for turning down the application are given.

3532 COMPUTER MODELLING OF PIPING COMPONENTS FOR TRANSIENT HYDRODYNAMIC-STRUCTURAL RESPONSE. A-Moneim, M.T.; Chang, Y.W.; Fistedis, S.H. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 49: No. 1, 175-186 (Jul 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, Calif., USA (15 - 19 Aug 1977).

The paper describes three new piping components that are coupled with existing pipe and elbow models so that transient hydrodynamic-structural response analysis of general piping systems is possible. The three models are: a generalized piping component model, a tee-branching junction model, and a surge tank model. Optional interior rigid wall simulation and heat exchanger tube bundle representation in the generalized piping component model makes possible its use in modelling valves, pipe expansions and reducers, and heat exchangers. Two-dimensional implicit continuous-fluid Eulerian finite difference technique is utilized in the hydrodynamics calculations. A convected coordinates finite element method is used in the structural analysis which considers both the membrane and bending strengths of the thin axisymmetric shells representing the walls.

3533 COMPARISON OF ICEPEL CODE PREDICTIONS WITH STRAIGHT FLEXIBLE PIPE EXPERIMENTS. A-Moneim, M.T.; Chang, Y.W. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 49: No. 1, 187-196 (Jul 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, Calif., USA (15 - 19 Aug 1977).

Experimental results on the pressure pulse propagation along plastically deforming pipes are interpreted. The tests are also analyzed by the ICEPEL piping code and the results compared with the experimental measurements of both pressure and circumferential strain histories in the pipe.

3534 COMPARISON OF FINITE ELEMENT CALCULATIONS WITH EXPERIMENTS FOR THE DYNAMIC RESPONSE OF LMFBR CORE SUBASSEMBLY DUCTS. Ash, J.E. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 49: No. 1, 79-92 (Jul 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, Calif., USA (15 - 19 Aug 1977).

The capability of a finite-element computer code STRAW to predict the structural response of LMFBR core subassembly hexagonal wrappers to postulated accidental local energy releases has been verified by comparisons with test data from out-of-pile experiments. The code was applied to the analysis of the dynamic response of hexcans subjected to internal pressure pulse loadings, and the computational results were correlated with data from a series of tests on individual hexcan sections. The pressure was generated by a PETN burning mixture. Two specified pressure pulse shapes were generated to simulate hypothetical conditions created by a fuel-coolant interaction resulting from a fuel pin failure.

3535 FINITE ELEMENT FORMULATION FOR THERMAL STRESS ANALYSIS OF THIN REACTOR STRUCTURES. Kulak, R.F.; Kennedy, J.M. (Argonne National Lab., Ill. (USA)); Belytschko, T.B. (Northwestern Univ., Evanston, Ill. (USA)). Dept. of Civil Engineering; Schoeberle, D.F. (Illinois Univ., Chicago (USA)). Nucl. Eng. Des.; 49: No. 1, 39-50 (Jul 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, Calif., USA (15 - 19 Aug 1977).

This paper describes the formulation of a finite-element procedure for the thermal stress analysis of thin wall reactor components. A general temperature-dependent constituent relationship is derived from a Gibbs potential function and a temperature-dependent yield surface. This form of constitutive relationship is applicable to problems of small strain. A similar form of a hypoelastic-plastic type is also developed for large strains. The variation of the yield surface with temperature is based upon a temperature-dependent, work-hardening model. The model uses a temperature-equivalent stress-plastic strain diagram which is generated from isothermal uniaxial stress-strain data.

3536 EFFECTS OF CRACKS IN IRRADIATION-EMBRITTLED HEXAGONAL SUBASSEMBLY DUCTS. Petroski, H.J.; Glazik, J.L. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 49: No. 1, 51-68 (Jul 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, Calif., USA (15 - 19 Aug 1977).

Plane-strain stress intensity factors for hexagonal subassembly ducts with a crack in a corner or midflat are presented in convenient graphical form for representative LMFBR hexcan dimensions. Calibrations based on several different models of the round hexcan corner and a finite element analysis of a uniformly pressurized hexcan with a corner crack corroborate each other. A weight function for this geometry is validated, and simple analytical tools are demonstrated to be adequate for parametric studies of cracked hexagonal ducts. The effects of different numbers of cracks, different locations for cracks, and different loading modes are discussed and some comments are made on the application of linear elastic fracture mechanics to cracked hexagonal ducts which have suffered a high degree of fast-neutron irradiation embrittlement.

- 3537 NUMERICAL METHOD FOR PREDICTING SEISMIC-INDUCED IMPACT BETWEEN SUBASSEMBLIES IN LMFBR CORES. Schreyer, H.L. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 49: No. 1, 69-78(Jul 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, Calif., USA (15 - 19 Aug 1977).

Structural problems that incorporate impact usually require small time steps for numerical integration and, consequently, solutions are very expensive. Since impact is one of the basic phenomena involved in any study of fast breeder reactor cores subjected to seismic disturbances, it is imperative that simplified models be introduced to make safety studies economically feasible. An approach is proposed whereby just the magnitude of each impulse and not the history of the impact force is determined for each impact. The consequence of the procedure is that the maximum time step is governed by system parameters and not by a detailed characteristic of the contact region such as an equivalent contact spring. The concepts are developed with the use of a single-degree-of-freedom model and the approach is applied to a three-assembly model of a reactor core.

- 3538 COMPARISON OF REXCO CODE PREDICTIONS WITH FLEXIBLE VESSEL EXPERIMENTS. Chang, Y.W.; Gvildys, J. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 49: No. 1, 131-144(Jul 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, Calif., USA (15 - 19 Aug 1977).

This paper deals with the REXCO code predictions of the flexible vessel tests. Comparisons are made with the data from experiments which were designed specifically for the purpose of code verification and/or modification. These experiments were performed with a well-defined and calibrated energy source in cylindrical vessel with precision tolerances and simple well-defined boundary conditions. The experimental data can thus be used as reliable test data for validation of computer codes, as well as of the modeling techniques used in the computer analysis. The inputs to the computer analysis are the vessel and core barrel dimensions and boundary conditions, the stress-strain relationships for the vessel and core barrel materials, the equation of state for the coolant, and the pressure-volume relationship of the energy source. The REXCO-predicted wall deformations, pressure loadings, and integrated impulses at various gauge positions are compared with the experimental data. Results of the comparisons are discussed.

- 3539 ANALYSIS OF LMFBR CONTAINMENT RESPONSE USING A MULTIFIELD IMPLICIT CONTINUOUS-FLUID EULERIAN CODE (MICE). Chu, H.Y.; Chang, Y.W. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 49: No. 1, 107-118(Jul 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, Calif., USA (15 - 19 Aug 1977).

An Eulerian computer code, MICE, for analyzing multifield-fluid flow problems in LMFBR containments is presented. The hydrodynamics of the MICE code is based upon the implicit multifield (IMF) method which includes the treatment of multifield fluids, the interpenetration of materials, the heat transfer, and the material phase changes. The finite-difference equations and the numerical techniques used in obtaining the equilibrium

pressures and in calculating the fluid-structure interactions are described in detail. Sample problems are given to illustrate the capabilities of the code.

- 3540 RESPONSE OF BREEDER REACTOR COMPONENTS AND SYSTEMS TO ABNORMAL EVENTS. Fistedis, S.H. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 49: No. 1, 3-15(Jul 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, Calif., USA (15 - 19 Aug 1977).

The paper contains a brief overview of applied mechanics as it is utilized in the safety and design of fast breeder nuclear reactors. It also attempts to consolidate and delineate the objectives of the following 13 specific papers of the same program, into an entity describing the present status and the remaining problems of this sector of nuclear technology.

- 3541 DYNAMIC STRUCTURAL RESPONSE OF LMFBR HEAD CLOSURES TO HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS. Kulak, R.F. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 49: No. 1, 119-129(Jul 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, Calif., USA (15 - 19 Aug 1977).

Presented here is an investigation of the dynamic structural response of the primary vessel's head closure to a hypothetical core disruptive accident (HCDA). Two head-closure designs were considered: the first represents a loop-type design and the second represents a pool-type design. Using representative configurations of liquid metal fast breeder reactors (LMFBR), independent models were used (1) to derive loading pressure histories and (2) to study the structural response of the head closures. Results for loading pressures, displacement histories, deformed profiles, stress magnitudes and plastically deformed regions are presented.

- 3542 ANALYSIS AND APPLICATION OF PRESTRESSED CONCRETE REACTOR VESSELS FOR LMFBR CONTAINMENT. Marchertas, A.H.; Fistedis, S.H. (Argonne National Lab., Ill. (USA)); Bazant, Z.P.; Belytschko, T.B. (Northwestern Univ., Evanston, Ill. (USA). Dept. of Civil Engineering). Nucl. Eng. Des.; 49: No. 1, 155-173(Jul 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, Calif., USA (15 - 19 Aug 1977).

An analytical model of a prestressed concrete reactor vessel (PCRV) for LMFBR and the associated finite element computer code, involving an explicit time integration procedure, is described. The model is axisymmetric and includes simulations of the tensile cracking of concrete, the reinforcement, and a prestressing capability. The tensile cracking of concrete and the steel reinforcement are both modeled as continuously distributed within the finite element. The stresses in the reinforcement and concrete are computed separately and combined to give an overall stress state of the composite material. Attention is given to the fact that cracks do not form instantaneously, but develop gradually.

- 3543 COMPARISON OF ICECO CODE PREDICTIONS WITH FLEXIBLE VESSEL EXPERIMENTS. Wang, C.Y. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 49: No. 1, 145-154(Jul 1978).

From 4. international conference on structural mechanics in reactor technology; San

Francisco, Calif., USA (15 - 19 Aug 1977).

Some comparisons of ICECO code predictions with experimental data concerning transient fluid-structure interaction are given. The test results are taken from flexible vessel experiments conducted by Stanford Research Institute under the direction of Argonne National Laboratory. Two different experiments are considered: one with a rigid core barrel, and one with a flexible core barrel. Both experiments are performed in simple reactor vessels with a well-defined energy source and simple boundary conditions. Correlations of pressures and impulses are made at all available gauge stations. The permanent deformations of the core barrel and the cylindrical vessels are compared with ICECO predictions. The effects of core barrel flexibility on the wave propagation and vessel deformation are also investigated. The agreement between the analysis and experiments is found to be quite good.

- 3544 ANALYSIS OF NONLINEAR FLUID-STRUCTURE INTERACTION IN LMFBR CONTAINMENT. Wang, C.Y.; Chang, Y.W.; Fistedis, S.H. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 49: No. 1, 93-105 (Jul 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, Calif., USA (15 - 19 Aug 1977).

A generalized Eulerian method has been incorporated into ICECO for analyzing the nonlinear fluid-structure interaction in the primary containment of an LMFBR, consisting of complicated structural components such as the radial shield, core barrel, core-support structure, and the primary vessel. The method employs a Poisson equation to determine the hydrodynamic pressure in the fluid region while using a relaxation equation to compute the pressure adjacent to the structure. A generalized coupling scheme is developed for treating the sliding condition at the fluid-structure interface, modeling the perforated structure, and analyzing the fluid motion at the geometrical discontinuities. Detailed formulations are given. Sample problems concerning wave propagation in a typical reactor containment are presented. It is shown from the results that this implicit, iterative method is unconditionally stable, and is especially suitable for excursions involving large material distortions.

- 3545 ABILITY TO CHALLENGE PARTIAL LICENCES ACCORDING TO THE ATOMIC LAW. REMARKS ON THE VERDICT OF THE HIGH COURT AT MUENSTER ON 16. JUNE 1978. Buedenbender, U. Energiewirtsch. Tagesfragen; 28: No. 7, 429-431 (Jul 1978). (In German).

The verdict of the High Court at Munster deserves full agreement. The verdict denied the legal interest of the plaintiff stopping the completion of the first partial licence of construction with convincing reasons, in spite of practically complete execution of the construction work contained in the first partial licence of construction, who had complained because of his legal conception of the relation of the first partial licence of construction to subsequent licences. The significance of this decisions extends beyond the concrete fast breeder project and even beyond legal processes in nuclear power station projects, to basic statements of the meaning of the first partial licence for subsequent licences.

- 3546 NUMERICAL STUDY OF BURNUP CHARACTERISTICS OF 300 MWE LMFBR BASED ON THREE-

DIMENSIONAL CALCULATION. Takeda, S. (Hitachi Ltd., Ibaraki (Japan). Energy Research Lab.). J. Nucl. Sci. Technol. (Tokyo); 15: No. 7, 502-513 (Jul 1978).

The burnup characteristics of a 300 MWe LMFBR are studied numerically based on three-dimensional calculations. The analysis covers the effects brought by differences in the assumed positions of the control rods and by changing the number of energy groups adopted for calculating the burnup reactivity loss and the detailed space distribution of major Pu isotopes after prescribed periods of irradiation. Results are also compared between three- and two-dimensional models used for the calculations. The results reveal significant influence of control rod position on the depletion and buildup of Pu isotopes not only in the core region but also in the blanket region. Burnup calculations based on two energy groups prove to provide a fairly good approximation of the results obtained with three groups; this is not the case with one-group calculations, so long as the conventional method is adopted for reducing the energy groups. The errors introduced by the approximative methods in estimating the changes brought to the amounts of Pu isotopes resulting from irradiation increases with the atomic weight, in the order ^{239}Pu , ^{240}Pu and ^{241}Pu . Hence for accurate estimation of breeding characteristics, it would be necessary to perform three-dimensional burnup calculations with account taken of the individual positions of control rods distributed in discrete channels over the cross sectional plane. The use of an adequate number of energy groups is also essential unless a suitable improved one-group method is adopted.

- 3547 CORRELATION MEASUREMENTS OF SODIUM FLOW RATE WITH MAGNETIC SENSORS. Kebabdz, B.V.; Krasnoyarov, N.V.; Adamovskii, L.A.; Golushko, V.V.; Sroelov, V.S. At. Energ. (USSR); 45: No. 1, 30-35 (Jul 1978). (In Russian).

The results of bench-mark experiments and those carried out at the BOR-60 reactor to measure the sodium coolant flow rate by a correlation method are presented. The method is based on detecting the eddy type flow hydraulic nonuniformities using magnetic flowmeters. The measurements were fulfilled in a broad range of flow rates ($G=10-10^4 \text{ m}^3/\text{h}$, $Re=2 \times 10^5-2 \times 10^7$). The measured and calculated mutual correlation functions are presented with parallel and perpendicular orientations of the flowmeters magnetic fields. A good accord is stated. Prerequisites to the arrangement of the measuring systems are formulated. As an important advantage of the correlation method a possibility of the flowmeter calibration in situ is highlighted.

- 3548 INCIPIENT BOILING OF SODIUM IN SEVEN-PIN BUNDLE UNDER FORCED CONVECTION CONDITIONS. Kikuchi, Y.; Haga, K. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). J. Nucl. Sci. Technol. (Tokyo); 15: No. 7, 533-542 (Jul 1978).

Experimental studies were carried out on incipient boiling of sodium flowing in an electrically heated seven-pin bundle under forced convection. In the first series of experiments temperature distributions in the bundle were measured under non-boiling conditions. The measured temperature distributions agreed fairly well with the calculation by the NORMAL code. In the second series of experiments incipient-boiling (IB) phenomena were investigated, with particular reference to the IB superheat and the voiding

patterns. The IB wall superheat decreased with increase in flow velocity. The observed coolant voiding was limited in the central subchannel because of the steep radial temperature gradient in the bundle. In order to describe this voiding process a two-dimensional voiding model was required.

- 3549 SUBASSEMBLY ANALYSIS. McAreavey, G. (UKAEA, Risley. Nuclear Power Development Establishment). Nucl. Energy; 17: No. 3, 217-224 (Jul 1978).

Papers on sub-assembly analysis presented at the SMIRT 4 conference are reviewed together with others from the session on local effects, all in relation to LMFBR fuel elements. The papers are indicative of some problems of most concern to LMFBR designs, in relation to clad and structural materials property changes due to fast reactor irradiation. The topics discussed are the interaction between bowing and diluting of sub-assemblies in relation to restrained cores, irradiation creep deformation of wrapper walls, dynamic loading of defective and embrittled ducts, and bowing of fuel pins. A number of conclusions are drawn.

- 3550 STUDY OF THE VERTICAL STRAIGHT TUBE TYPE STEAM GENERATOR FOR LMFBR, 1. TEST RIG OF A SODIUM-HEATED STEAM GENERATOR AND EXPERIMENTAL RESULTS OF DRYOUT. Roko, K.; Shiraha, M.; Yoshizaki, A.; (Kawasaki Heavy Industries Ltd., Akashi, Hyogo (Japan). Technical Inst.); Takitani, K.; Sakano, K. Kawasaki Giho; No. 67, 8-14 (Jul 1978). (In Japanese).

This paper outlines a 0.1 MWt test rig of a LMFBR (Liquid Metal Fast Breeder Reactor) vertical straight tube type sodium-heated steam generator which has been built at KHI's Technical Institute for studying various aspects of the steam generator. It also presents the experimental results of dryout quality with special attention given to heat transfer characteristics. The experiments covered a steam pressure range of 52 to 153 kg/cm² and a mass velocity range of 250 to 550 kg/m²s. A new correlation for the dryout quality was proposed in terms of the steam pressure and the mass velocity. The comparison of the correlation with the experimental data from the above-mentioned test and the preliminary Joule-heated test verified that it was applicable to a broad steam pressure range of 18 to 153 kg/cm² and a mass velocity range of 160 to 550 kg/m²s. The R.M.S. (root mean square) error of the correlation was 0.0232.

- 3551 COMPUTER MODELS FOR SUBASSEMBLY SIMULATION. Belytschko, T.B. (Northwestern Univ., Evanston, Ill. (USA). Dept. of Civil Engineering); Kennedy, J.M. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 49: No. 1, 17-38 (Jul 1978).

Two-dimensional finite element models for the treatment of the nonlinear, transient response of fluids and structures are described. The fluid description is quasi-Eulerian, so that the mesh can move independently of the material and it includes a new finite element upwinding scheme. The structural description is based on a corotational formulation in which the coordinate system is embedded in the elements, and which is applicable to arbitrarily large rotations. The interface between the fluid and structure permits relative sliding, but because of the quasi-Eulerian fluid description the nodes of the fluid and structure can be allowed to remain contiguous. Modelling procedures for treating the various aspects of subassemblies, such as the narrow fluid channels, the fuel

bundles which are immersed in the coolant, and the axial flow, are developed. Calculations are made for a symmetric seven-subassembly cluster and compared to experimental results. In addition, the application to a 19-subassembly cluster is described.

- 3552 WASTAGE PHENOMENA BY SMALL LEAK SODIUM-WATER REACTION. II. PARAMETERS EFFECT. Nei, H.; Hashiguchi, K. (Toshiba Corp., Kawasaki, Kanagawa (Japan)). Nippon Kikai Gakkai Ronbunshu; 44: No. 383, 2439-2448 (Jul 1978). (In Japanese).

The wastage phenomena due to sodium-water reaction were studied both experimentally and theoretically. An equation for the wastage rate, which is defined as the value of maximum wastage depth divided by steam injection time, was derived semi-analytically. First, an empirical formula for the wastage rate was derived as a function of sodium temperature and steam leak rate. Secondly, the leak rate corresponding to the maximum wastage rate was replaced by an equation derived from the flame model theory, and the semi-analytical equation for wastage rate was finally obtained as a function of many parameters such as nozzle diameter, nozzle-target distance, steam pressure, etc. The accuracy of this equation was verified by comparing with the results of experiments by the authors and other researchers. This paper explains the derivation of the equation together with its experimental background, and evaluates the applicability of this equation.

- 3553 FAST REACTOR FUEL PIN PERFORMANCE. Matthews, J.R. (UKAEA, Harwell. Atomic Energy Research Establishment). Nucl. Energy; 17: No. 3, 211-216 (Jul 1978).

Papers on fast reactor fuel performance presented at the SMIRT 4 conference and at the post-conference seminar on fuel element modelling are reviewed. The principal computer codes used for analysis of pin behaviour are discussed. Other main topics are cladding swelling, fuel cracking, studies on fuel pin failures and accident analysis of fuel cladding.

- 3554 REVISION OF THE SECOND BASIC PLANS OF POWER REACTOR DEVELOPMENT IN POWER REACTOR AND NUCLEAR FUEL DEVELOPMENT CORPORATION. Genshiryoku Iinkai Geppo; 23: No. 3, 9-11 (Jul 1978). (In Japanese).

Revision of the second basic plans concerning power reactor development in PNC (Power Reactor and Nuclear Fuel Development Corporation) is presented. (1) Fast breeder reactors: as for the experimental fast breeder reactor, after reaching the criticality, the power is raised to 50 MW thermal output within fiscal 1978. The prototype fast breeder reactor is intended for the electric output of 200 MW to 300 MW, using mixed plutonium/uranium oxide fuel. Along the above lines, research and development will be carried out on reactor physics, sodium technology, machinery and parts, nuclear fuel, etc. (2) Advanced thermal reactor: The prototype advanced thermal reactor, with initial fuel primarily of slightly enriched uranium and heavy water moderation and boiling water cooling, of 165 MW electric output, is brought to its normal operation by the end of fiscal 1978. Along the above lines, research and development will be carried out on reactor physics, machinery and parts, nuclear fuel, etc.

- 3555 (DOE-tr--176) EXPERIMENTAL

INVESTIGATION OF THE TURBULENT FLOW THROUGH A WALL SUBCHANNEL OF A ROD BUNDLE. Rehme, K. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). 30 Jun 1978. Translation of KFK--2441, April 1977 114p. Dep. NTIS, PC A06/MF A01.

Work performed under United States--W. Germany Fast Reactor Exchange Program.

An experimental investigation was carried out to obtain reliable information on the transport properties of turbulent flow through subchannels of rod bundles. Detailed data were obtained on the distributions of the average-flow velocity, the turbulence intensities in all directions and, thus, the kinetic energy of turbulence, the shear stresses in the directions normal and parallel to the walls, and the wall shear stresses for a wall subchannel of a rod bundle of four parallel rods. The pitch-to-diameter ratio of the rods, which was equal to the wall-to-diameter ratio, was 1.07, and the Reynolds number for the investigation was $Re = 8.7 \times 10^4$. On the basis of the data measured, the eddy viscosities in the directions normal to and parallel to the walls were calculated. Thus, detailed data on the eddy viscosities in directions parallel to the walls in rod bundles were obtained for the first time. The results of the measurements were compared with the results of the VELASCO computations.

3556 FISSION PRODUCT ARRANGEMENT OF A FAST NEUTRON REACTOR. Iliunin, V.G.; Murogov, V.M.; Troianov, M.F.; Rinieiskii, A.A.; Schmelev, A.N. (to Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,656,841/A/. 22 Jun 1978. 15p. (In German).

In fast breeder reactors, the large spatial share of the structural materials means a worsening of the neutron balance. By reducing the hydraulic resistance of the fuel element arrangement, the balance is partially improved. The fuel elements are arranged so that the total height without shortening the fuel element length is considerably reduced.

3557 NUCLEAR REACTOR REFUELING SYSTEM. Wade, E.E. (to Westinghouse Electric Corp.). US Patent 4,096,031. 20 Jun 1978. Filed date 22 Jul 1976. 10p.

A system for transferring fuel assemblies between a nuclear reactor core and a fuel storage area while the fuel assemblies remain completely submerged in a continuous body of coolant is described. The system comprises an in-vessel fuel transfer machine located inside the reactor vessel and an ex-vessel fuel transfer machine located in a fuel storage tank. The in-vessel fuel transfer machine comprises two independently rotatable frames with a pivotable fuel transfer apparatus disposed on the lower rotatable frame. The ex-vessel fuel transfer machine comprises one frame with a pivotable fuel transfer apparatus disposed thereon. The pivotable apparatuses are capable of being aligned with each other to transfer a fuel assembly between the reactor vessel and fuel storage tank while the fuel assembly remains completely submerged in a continuous body of coolant. 9 claims, 7 figures.

3558 MONITORING DEVICE FOR NUCLEAR REACTOR CONTROL RODS. Blanc, R.; Jacquelin, R. (to Commissariat a l'Energie Atomique). US Patent 4,094,369. 13 Jun 1978. Priority date 8 Oct 1974, France, 8p.

Weighing of each control rod and at least

the moving portion of the control rod drive mechanism is carried out in continuous operation by means of an assembly comprising at least one strain gage mounted in a circuit for measuring the electrical resistance of the gage according to the change in length which is a function of the weight being measured.

3559 AUTOMATIC CONTROL MECHANISM FOR NUCLEAR REACTOR SCRAM. Giuggio, N.; Croissant Noyes, R.; Uz Zaman, S. (to Combustion Engineering, Inc., New York; USA). French Patent 2,394,150/A/. 9 Jun 1978. 26p. (In French).

A mechanism is described in which the mean temperature of the coolant on leaving the core of a liquid cooled nuclear reactor is detected and which allows a poison to fall quickly into the core of the reactor when a critical temperature is reached. The reactor comprises a safety pipe located in the core and rising above it, a vertical flow path for the reactor coolant passing through the pipe and a poison bundle absorbing the neutrons, retained by a supporting system. Several fuel rods are located in the flow path of the coolant, a temperature sensitive control device is located in the flow path near the upper end of the fuel rods, a retaining device is fitted between the poison bundle support system and the control system so as to release the poison bundle when the control system makes a preset movement.

3560 SWELLING ANALYSIS OF HIGHLY-RATED MX-TYPE LMFBR FUELS. II. MICROSCOPIC SWELLING BEHAVIOR. Ronchi, C.; Ray, I.L.F.; Thiele, H.; Van De Laar, J. (Comm of the Eur Communities, Eur Inst for Transuranium Elem, Jt Res Cent, Karlsruhe, Ger). J. Nucl. Mater.; No. 2, 193-211(2 Jun 1978).

Fission gas bubbles in these compounds grow to large sizes if the in-pile temperature rises above a critical value (swelling critical temperature $T_{sub c}$). A comparison has been made of the swelling rates in fuels of different composition, showing that $T_{sub c}$ increases from carbides to nitrides. In fuels subjected to in-pile restructuring (highly rated He-bonded pins) microscopic swelling is affected by pore and grain boundary migration. 16 refs.

3561 COMPOSITION OF THE CORROSION PRODUCT OXIDE PHASES IN THE GAPS OF THREE LMFBR-TYPE MIXED OXIDE FUEL PINS. Walker, C.T. (Jt Res Cent, Eur Inst for Transuranium Elem, Karlsruhe, Ger). J. Nucl. Mater.; 74: No. 2, 358-362(2 Jun 1978).

Full analyses (including oxygen) are presented for the gray and light-gray corrosion product phases in the gap of pin section AU07-10. The findings support the proposal that chromium oxide is the principal constituent of the phases. 9 refs.

3562 FAST BREEDER REACTORS. (to Nuclear Power Co. (Whetstone) Ltd. (UK)). French Patent 2,393,405/A/. 2 Jun 1978. 13p. (In French).

The subject of this invention is a liquid metal cooled nuclear reactor construction in which a concrete pit is lagged to protect it from the heat radiated from the reactor in normal operation but in which the efficiency of the lagging is reduced in case of emergency to allow the excess heat generated by the reactor to be dissipated throughout the pit. The lagging is in two layers, the first covering the internal surface of the pit wall is impermeable to the liquid metal, whilst the

second layer over the first is permeable.

- 3563 NUCLEAR REACTOR. (to Nuclear Power Co. (Whetstone) Ltd. (UK)). French Patent 2,393,406/A/. 2 Jun 1978. 8p. (In French).

A nuclear reactor is described comprising a fuel assembly and components housed in a pit and a chamber located above the pit containing a device for handling the components of the reactor, passages being provided in the roof of the pit for transferring the reactor components.

- 3564 FUEL ASSEMBLY FOR FAST BREEDER LIQUID METAL COOLED REACTOR. (to Nuclear Power Co. (Whetstone) Ltd. (UK)). French Patent 2,393,402/A/. 2 Jun 1978. 9p. (In French).

The description is given of a fuel assembly for a fast breeder liquid metal cooled reactor comprising a bundle of fuel pins spaced out in a sheath with a pointed extension for fitting into a supporting housing in the fuel assembly, the feature of this rod being that the sheath is articulated at the pointed extension.

- 3565 METHOD OF REDUCING THE HAZARD WHICH MAY OCCUR AS A CONSEQUENCE OF A REACTOR CORE MELTDOWN. Donne, M.D.; Dorner, S.; Schumacher, G. (to Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.); Deutsches Patentamt, Muenchen (Germany, F.R.)). German(FRG) Patent 2,653,258/A/. 1 Jun 1978. 14p. (In German).

The core melt resulting from a meltdown accident of a GFB, LWR or LMFBR is collected by a core catcher from graphite placed below the core. The core melt is penetrating step by step into a borate store in the collecting vessel and is dissolving in it. Therefore the borate at the same time will absorb the decay heat. In order to remove the solidified and cooled down melted mass water is applied eliminating the borate. The remaining oxide state of the powdery core is sucked off again from the core catcher together with the water. The borate store (e.g. alkali borate) itself consists of separate layers with shaped parts, the coverings of which are made of steel, iron, cast iron, nickel, iron or nickel alloys, ceramic material or glass.

- 3566 MEASURING PROBE AND METHOD FOR DETERMINING THE OXYGEN CONTENT IN GASES, VAPORS, AND LIQUIDS, ESPECIALLY IN LIQUID METALS. Sundermann, H.; Andrae, U. (to Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.); Deutsches Patentamt, Muenchen (Germany, F.R.)). German(FRG) Patent 2,061,599/B/. 1 Jun 1978. 4p. (In German).

The invention is concerned with the improvement of the measuring probe described in the main patent no. 1798002 with which the oxygen content in liquid metals, e.g. Na, is to be determined. In order to avoid the glass stopper shutting off the reference space having to be ground out it is proposed to connect the solid electrolyte firmly and hermetically with a metallic mounting support (e.g. Fe-Co-Ni alloy), having the same thermal coefficient of expansion as the solid electrolyte (e.g. zirconium dioxide stabilized with yttrium oxide or thorium dioxide). Further details of the design are very explicitly described.

- 3567 EFFECTS OF INTERNAL CIRCULATION VELOCITY AND NONCONDENSABLE GAS ON VAPOR CONDENSATION

FROM A RISING BUBBLE. Gezisik, M.N. (North Carolina State Univ., Raleigh); Kress, T.S. Nucl. Sci. Eng.; 66: No. 3, 397-405(Jun 1978).

The effects of internal circulation velocity and the presence of noncondensable gas on vapor removal rate by condensation from a rising large vapor-gas bubble produced in a hypothetical core disruptive accident are investigated by solving the resulting transient heat and mass transfer problem of turbulent flow. Sample calculations are performed for the condensation of UO_2 and sodium vapors containing noncondensable fission gases. The time-averaged condensation heat transfer coefficients are presented for the condensation of UO_2 and sodium vapors for different internal circulation velocities and the concentration of noncondensable gas.

- 3568 EQUILIBRIUM THERMODYNAMICS OF URANIUM--PLUTONIUM CARBIDE LMFBR FUEL. Besmann, T.M. (Oak Ridge National Lab., TN); Potter, P.E. Trans. Am. Nucl. Soc.; 28: 217-219(Jun 1978). From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

See CONF-780622--.
URANIUM CARBIDES; PLUTONIUM CARBIDES; PHASE STUDIES; EQUILIBRIUM; THERMODYNAMICS; LMFBR TYPE REACTORS

- 3569 PREIRRADIATION OF LMFBR FRESH FUEL ASSEMBLIES TO IMPROVE PROLIFERATION RESISTANCE. Talwar, B.; Deane, N.A.; Bailey, H.S. (General Electric Co., Sunnyvale, CA). Trans. Am. Nucl. Soc.; 28: 375(Jun 1978).

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
FUEL ASSEMBLIES; IRRADIATION; SAFEGUARDS; LMFBR TYPE REACTORS

- 3570 USE OF HODOSCOPES TO OBTAIN TOMOGRAPHIC RESOLUTION OF FUEL MOTION. Person, L.W.; Klickman, A.E.; Rhodes, E. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 28: 127-128(Jun 1978).

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR. LMFBR TYPE REACTORS; FUEL PINS; REACTOR ACCIDENTS; INSPECTION; HODOSCOPES; PERFORMANCE

- 3571 INTENTIONAL STEAM TUBE FAILURE IN AN LMFBR-TYPE STEAM GENERATOR. Neely, H.H. (Atomics International, Canoga Park, CA); DeBear, W.S.; Miller, A.E.; Regimbal, J.J. Trans. Am. Nucl. Soc.; 28: 184-185(Jun 1978).

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; TUBES; FAILURES; SODIUM; CORROSIVE EFFECTS; STEAM GENERATORS

- 3572 EFFECTS OF STOICHIOMETRY ON CLADDING REACTION IN MIXED-OXIDE FUEL AT HIGH BURNUP. Lawrence, L.A.; Weber, J.W.; Devary, J.L. (Westinghouse Hanford Co., Richland, WA).

Trans. Am. Nucl. Soc.; 28: 210-212(Jun 1978). From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; FUEL-CLADDING INTERACTIONS; STOICHIOMETRY; URANIUM DIOXIDE; PLUTONIUM DIOXIDE; STAINLESS STEEL-316

- 3573 RESULTS FROM EXPERIMENT F11A AFTER INCURRING CLADDING BREACH IN EBR-II. Rosa, J.M.; Hilbert, R.F. (General Electric Co.,

- Sunnyvale, CA). Trans. Am. Nucl. Soc.; 28: 214-216(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
EBR-2 REACTOR;FUEL PINS;FUEL ELEMENT FAILURE; RUPTURES
- 3574 PERFORMANCE OF EBR-II MARK-II METALLIC DRIVER FUEL TO 10 AT.% BURNUP. Einzinger, R.E.; Seidel, B.R. (Argonne National Lab., Idaho Falls). Trans. Am. Nucl. Soc.; 28: 219-220(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
EBR-2 REACTOR;FUEL PINS;PERFORMANCE;PHYSICAL RADIATION EFFECTS;BURNUP
- 3575 EXPERIMENTAL CONFIRMATION OF THE DESIGN TO MINIMIZE VIBRATION AND WEAR IN 61-PIN WIRE-SPACED EBR-II SUBASSEMBLIES. Fukuda, S. (Westinghouse Hanford Co., Richland, WA). Trans. Am. Nucl. Soc.; 28: 222-223(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
EBR-2 REACTOR;FUEL ASSEMBLIES;WEAR; MECHANICAL VIBRATIONS;SPACERS
- 3576 RADIAL DISTRIBUTION OF RETAINED FISSION GAS IN IRRADIATED MIXED-OXIDE FUEL. Randklev, E.H. (Westinghouse Hanford Co., Richland, WA). Trans. Am. Nucl. Soc.; 28: 234-236(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS;SOLID FUELS;FISSION PRODUCTS;GASES;DISTRIBUTION;MICROSTRUCTURE
- 3577 STATISTICAL ANALYSIS OF INTERACTING COOLANT CHANNELS. May, R.S. (Univ. of Illinois, Urbana); Hagan, W.K. (Science Applications, Inc., La Jolla, CA). Trans. Am. Nucl. Soc.; 28: 278-279(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR. LMFBR TYPE REACTORS;REACTOR COOLING SYSTEMS;TEMPERATURE GRADIENTS;STATISTICAL MODELS
- 3578 EFFECTS OF MATERIAL PROPERTIES ON THE DYNAMIC RESPONSE OF PRESSURE-LOADED CORE SUBASSEMBLY WRAPPERS. Ash, J.E. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 28: 464-465(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS;FUEL ELEMENT CLUSTERS; SHROUDS;DEFORMATION;PRESSURE GRADIENTS;STRAINS; YIELD STRENGTH;STAINLESS STEEL-316
- 3579 TURBULENT FLOW SPLIT EXPERIMENT AND MODEL FOR WIRE-WRAPPED ASSEMBLIES. Chiu, C.; Tedreas, N.E.; Rohsenow, W.M. (Massachusetts Inst. of Tech., Cambridge). Trans. Am. Nucl. Soc.; 28: 536-537(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS;FUEL ELEMENT CLUSTERS; FLUID FLOW;TURBULENT FLOW
- 3580 THERMAL-HYDRAULIC ANALYSIS OF A WIRE-
- WRAPPED 19-ROD BUNDLE WITH EDGE BLOCKAGE. Lin, E.I.H.; Sha, W.T. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 28: 539-540(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS;FUEL ELEMENT CLUSTERS; HYDRAULICS;HEAT TRANSFER;FLUID FLOW;FLOW BLOCKAGE;T CODES;COMPUTER CALCULATIONS
- 3581 THREE-DIMENSIONAL NUMERICAL SIMULATION OF A 19-PIN LMFBR FUEL ASSEMBLY IN A HEXAGONAL DUCT. Domanus, H.M.; Schmitt, R.C.; Sha, W.T. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 28: 540-543(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS;FUEL ASSEMBLIES;FLUID FLOW;SHROUDS;THREE-DIMENSIONAL CALCULATIONS; VELOCITY
- 3582 THERMAL RADIATION TRANSFER THROUGH LMFBR DECK INSULATION. Shiu, K.; Jones, B.G. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 28: 549-551(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS;THERMAL SHIELDS;HEAT TRANSFER
- 3583 LMFBR VENTED ASSEMBLY HYDRAULIC MODELING AND TESTING. Wang, J.C.F. (General Electric Co., Schenectady, NY); Dubberley, A.E. Trans. Am. Nucl. Soc.; 28: 551-552(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS;FUEL ASSEMBLIES; HYDRAULICS;VENTS;FLOW MODELS
- 3584 PREDICTION OF THE THERMAL CONDUCTIVITY OF SPHERE-PAC FUEL. Peddicord, K.L.; Ades, M. (Oregon State Univ., Corvallis). Trans. Am. Nucl. Soc.; 28: 552-553(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS;THERMAL CONDUCTIVITY; SOLID FUELS;TWO-DIMENSIONAL CALCULATIONS
- 3585 DESIGN BASIS AND FTR TESTING OF AN ADVANCED WIRE-WRAP DRIVER FUEL ASSEMBLY. Kaplan, S.; Feerick, B.T.; Sim, R.G. (General Electric Co., Sunnyvale, CA). Trans. Am. Nucl. Soc.; 28: 559-560(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR. LMFBR TYPE REACTORS;FUEL ASSEMBLIES; SPECIFICATIONS;PERFORMANCE TESTING;HYDRAULICS; HEAT TRANSFER
- 3586 LMFBR CORE THERMAL-HYDRAULIC ANALYSIS ACCOUNTING FOR INTERASSEMBLY HEAT TRANSFER. Carelli, M.D.; Bach, C.W. (Westinghouse Advanced Reactors Div., Madison, PA). Trans. Am. Nucl. Soc.; 28: 560-562(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS;FUEL ASSEMBLIES; HYDRAULICS;HEAT TRANSFER;SPECIFICATIONS; COMPUTER CODES;COMPUTER CALCULATIONS

- 3587 PRACTICAL MODEL OF INTERASSEMBLY HEAT TRANSFER IN LMFBR ASSEMBLIES. Wei, J.P. (General Electric Co., Sunnyvale, CA). Trans. Am. Nucl. Soc.; 28: 562-563 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; FUEL ASSEMBLIES; HEAT TRANSFER; HYDRAULICS; COMPUTER CALCULATIONS; MATHEMATICAL MODELS
- 3588 BREEDING PERFORMANCE POTENTIAL FOR (U--TH) OXIDE, CARBIDE, NITRIDE, AND METAL-FUELED LMFBRs. Turski, R.B.; Lam, P.S.K.; Barthold, W.P. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 28: 563-565 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORES; REACTOR KINETICS; THORIUM CYCLE; BREEDING RATIO; COMPARATIVE EVALUATIONS
- 3589 RECOMMENDED CRBRP RELOAD CORE DESIGNS USING (U--PU)C FUEL. Vitti, J.A.; Galluzzo, N.G.; Brittingham, J.C. (Atomics International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 28: 565-566 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
CLINCH RIVER BREEDER REACTOR; REACTOR CORES; FUEL ASSEMBLIES; PLUTONIUM CARBIDES; URANIUM CARBIDES; REACTOR KINETICS; REACTIVITY WORTHS
- 3590 AXIAL PARFAIT LMFBR HETEROGENEOUS CORE DESIGNS. Naser, J.A.; Cokmez, S.; Sehgal, B.R. (Electric Power Research Inst., Palo Alto, CA). Trans. Am. Nucl. Soc.; 28: 566-567 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORES; SPECIFICATIONS; BREEDING RATIO; REACTOR KINETICS; REACTIVITY WORTHS; COMPUTER CALCULATIONS; COMPARATIVE EVALUATIONS; POWER DENSITY
- 3591 UNIQUE THERMAL AND HYDRAULIC FEATURES OF THE LIGHT-WATER BREEDER REACTOR. McWilliams, K.D.; Turner, J.R.; Stout, J.W.; Lerner, S. (Westinghouse Electric Corp., West Mifflin, PA). Trans. Am. Nucl. Soc.; 28: 575-576 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
SHIPPINGPORT REACTOR; LMFBR TYPE REACTORS; REACTOR CORES; HYDRAULICS; HEAT TRANSFER; FUEL ASSEMBLIES; CONTROL ELEMENTS
- 3592 EFFECT OF ALTERNATIVE FUEL CYCLES ON THE CLINCH RIVER BREEDER REACTOR. Dickson, P.W. Jr. (Westinghouse Advanced Reactors Div., Madison, PA). Trans. Am. Nucl. Soc.; 28: 577-578 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
CLINCH RIVER BREEDER REACTOR; FUEL CYCLE; PLANNING; THORIUM CYCLE
- 3593 EFFECT OF ALTERNATIVE FUEL CYCLES ON LMFBR DESIGN. Bailey, H.S. (General Electric Co., Sunnyvale, CA). Trans. Am. Nucl. Soc.; 28: 578-580 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
- See CONF-780622--.
LMFBR TYPE REACTORS; FUEL CYCLE; PLUTONIUM DIOXIDE; URANIUM DIOXIDE; THORIUM OXIDES; FUEL ASSEMBLIES; SPECIFICATIONS; FEASIBILITY STUDIES
- 3594 MESHLESS COLD TRAP: A PERFORMANCE EVALUATION. Goodman, L. (Atomics International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 28: 607 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR. LMFBR TYPE REACTORS; COLD TRAPS; PERFORMANCE TESTING
- 3595 LMFBR DESIGN WITH THE COMPOSITE FUEL ASSEMBLY. Cappiello, M.W.; Marr, D.R. (Westinghouse Hanford Co., Richland, WA). Trans. Am. Nucl. Soc.; 28: 608 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; FUEL ASSEMBLIES; SPECIFICATIONS; POWER DENSITY; HYDRAULICS; HEAT TRANSFER
- 3596 EFFECT OF STEAM CYCLE ON THE TRANSIENT BEHAVIOR OF LMFBRs. Adkins, C.R.; Gross, M.C.; Noe, D.E. (Westinghouse Advanced Reactors Div., Madison, PA). Trans. Am. Nucl. Soc.; 28: 608-610 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; STEAM SYSTEMS; TRANSIENTS; TEMPERATURE GRADIENTS; STEAM GENERATORS; THERMAL SHOCK
- 3597 CONSIDERATION OF FUEL MOTION POTENTIAL DUE TO VOLATILIZATION OF METALLIC FISSION PRODUCT INCLUSIONS. Sasa, P. (Univ. of New Mexico, Albuquerque); Cronenberg, A.; Stevenson, M.G. Trans. Am. Nucl. Soc.; 28: 426-428 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; FUEL RODS; FISSION PRODUCTS; EVAPORATION
- 3598 VIBRATION STUDIES OF PERFORATED PLATES ON CRBRP MIXING-TEE MODEL. Chung, H.; Hallie, H. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 28: 434-435 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
CLINCH RIVER BREEDER REACTOR; REACTOR COOLING SYSTEMS; MECHANICAL VIBRATIONS; PIPE FITTINGS; FLUID FLOW
- 3599 CREEP SPECIMEN CONTAINER AND EBR-II IRRADIATION SUBASSEMBLY DESIGN. Seim, O.S.; Wendte, J.C.; Feldman, E.E. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 28: 193-194 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
EBR-2 REACTOR; IRRADIATION CAPSULES; DESIGN; FUEL ELEMENT CLUSTERS; CREEP; PHYSICAL RADIATION EFFECTS
- 3600 THERMAL-HYDRAULIC ANALYSIS OF THE AIR DUMP HEAT EXCHANGERS IN THE FAST FLUX TEST FACILITY. Perkins, K.R.; Chen, L.; Bari, R.A. (Brookhaven National Lab., Upton, NY). Trans.

- Am. Nucl. Soc.; 28: 545-547 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
FFTF REACTOR; HEAT EXCHANGERS; HEAT TRANSFER;
HYDRAULICS; REACTOR COOLING SYSTEMS
- 3601 SOME BASIC RESEARCH NEEDS RELATED TO LWR
AND FBR SAFETY. Duffey, R.B. (Electric
Power Research Inst., Palo Alto, CA). Trans. Am.
Nucl. Soc.; 28: 55-56 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; BWR TYPE REACTORS; PWR
TYPE REACTORS; REACTOR SAFETY; RESEARCH PROGRAMS;
HYDRAULICS; HEAT TRANSFER
- 3602 MODELING BREEDER REACTOR TRANSIENT
FISSION GAS BEHAVIOR: BUBBLE EQUILIBRATION
EFFECTS. Gruber, E.E. (Argonne National
Lab., IL). Trans. Am. Nucl. Soc.; 28: 236-
237 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; TRANSIENTS; FISSION
PRODUCTS; FUEL PINS; BUBBLES; MATHEMATICAL MODELS
- 3603 MODELING OF THE EFFECTS OF GRAIN-
BOUNDARY FISSION GAS ON TRANSIENT FUEL
BEHAVIOR. DiMelfi, R.J.; Deitrich, L.W.
(Argonne National Lab., IL). Trans. Am. Nucl.
Soc.; 28: 237-239 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; TRANSIENTS; FUEL PINS;
FISSION PRODUCTS; GRAIN BOUNDARIES; GASES; BUBBLES;
MATHEMATICAL MODELS
- 3604 ROLE OF FISSION GAS AND FUEL MELTING ON
TRANSIENT FUEL RESPONSE. Bandyopadhyah, G.;
Buzzell, J.A. (Argonne National Lab., IL).
Trans. Am. Nucl. Soc.; 28: 241-243 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; TRANSIENTS; FUEL ELEMENTS;
MELTDOWN; FISSION PRODUCTS; GASES
- 3605 MODELING AND NUMERICAL METHODS OF
TWOPOOL. Sienicki, J.J.; Abramson, P.B.
(Argonne National Lab., IL). Trans. Am. Nucl.
Soc.; 28: 275-276 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR, LMFBR TYPE REACTORS; REACTOR CORE
DISRUPTION; THERMODYNAMICS; T CODES; HYDRODYNAMICS
- 3606 VALUE OF DISTRIBUTED PARTICLE IN CELL
TECHNIQUES. Abramson, P.B.; Sienicki, J.J.
(Argonne National Lab., IL). Trans. Am. Nucl.
Soc.; 28: 277-278 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR, LMFBR TYPE REACTORS; REACTOR CORE
DISRUPTION; HYDRODYNAMICS; MATHEMATICAL MODELS
- 3607 FIRST VISUAL IN-PILE FUEL DISRUPTION
EXPERIMENTS. Cano, G.L. (Sandia Labs.,
Albuquerque, NM). Trans. Am. Nucl. Soc.; 28:
437 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; LOSS OF FLOW; REACTOR
CORE DISRUPTION; MELTDOWN
- 3608 ANALYSIS OF IN-PILE FUEL DISRUPTION
EXPERIMENTS. Ostensen, R.W.; Young, M.F.
(Sandia Labs., Albuquerque, NM). Trans. Am.
Nucl. Soc.; 28: 437-438 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION;
LOSS OF FLOW; MELTDOWN
- 3609 SAS3A SIMULATIONS FOR A 19-PIN DRYOUT
EXPERIMENT. Han, J.T.; Tomlinson, E.T. (Oak
Ridge National Lab., TN). Trans. Am. Nucl.
Soc.; 28: 438-440 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; FUEL ELEMENT CLUSTERS;
DRYOUT; S CODES; COMPUTER CALCULATIONS
- 3610 SIMPLE TRANSIENT ANALYSIS OF THE THORS
BUNDLE 6A BOILING TEST RESULTS. Garrison,
P.W. (Oak Ridge National Lab., TN). Trans.
Am. Nucl. Soc.; 28: 440-442 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; FUEL ELEMENT CLUSTERS;
BOILING; HYDRAULICS; HEAT TRANSFER; ONE-
DIMENSIONAL CALCULATIONS
- 3611 EVALUATION OF INCOHERENCY EFFECT ON
CLADDING MOTION IN R-SERIES TEST. Ishii, M.;
Chen, W.L. (Argonne National Lab., IL).
Trans. Am. Nucl. Soc.; 28: 442-443 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; FUEL ELEMENTS; MELTDOWN;
BOILING; HEAT TRANSFER
- 3612 SODIUM VOIDING DYNAMICS AND CLADDING
MOTION IN A 37-PIN FUEL ASSEMBLY DURING AN LOF
TRANSIENT. Hwang, I.T.; Kuzay, T.M.; Marr,
W.W.; Miles, K.J. (Argonne National Lab.,
IL). Trans. Am. Nucl. Soc.; 28: 443-445 (Jun
1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; LOSS OF FLOW; FUEL
ASSEMBLIES; BOILING; MELTDOWN; FUEL CANS
- 3613 TWO-PHASE MOMENTUM EXCHANGE EXPERIMENT
AND ANALYSIS. Rexroth, P.E.; Starkovich, V.
(Los Alamos Scientific Lab., NM). Trans. Am.
Nucl. Soc.; 28: 445-446 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION;
HYDRODYNAMICS; MATHEMATICAL MODELS; TWO-PHASE
FLOW; DRAG
- 3614 SURFACE ABLATION IN THE IMPINGEMENT
REGION OF A LIQUID JET. Swedish, M.J.;
Epstein, M.; Linehan, J.H.; Lambert, G.A.;
Hauser, G.M.; Stachyra, L.J. (Argonne
National Lab., IL). Trans. Am. Nucl. Soc.; 28:
446-447 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.

- LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION;
MELTDOWN; SIMULATION; JETS
- 3615 ANALYSIS OF IN-PILE MFCI EXPERIMENTS
WITH CARBIDE FUEL. Henry, T.P.; Siegmann,
E.R.; Kennedy, M.F. (Combustion Engineering,
Inc., Windsor, CT). Trans. Am. Nucl. Soc.; 28:
452-453 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; MELTDOWN; FUEL-COOLANT
INTERACTIONS; URANIUM CARBIDES; PRESSURE
GRADIENTS
- 3616 DRYOUT HEAT FLUX IN DEBRIS BEDS
CONTAINING NONHEATED CONSTITUENTS AND PARTICLES
OF DIFFERENT SIZES. Dhir, V.K.; Catton, I.
(Univ. of California, Los Angeles). Trans.
Am. Nucl. Soc.; 28: 454-455 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; AFTER-HEAT
REMOVAL; DRYOUT; HEAT FLOW; SIMULATION; REACTOR
CORE DISRUPTION
- 3617 STUDY OF HEAT TRANSFER ASSOCIATED WITH
MELTING OF HORIZONTAL SUBSTRATE. Taghavi-
Tafreshi, K.; Dhir, V.K.; Catton, I. (Univ.
of California, Los Angeles). Trans. Am. Nucl.
Soc.; 28: 455-456 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; MELTDOWN; REACTOR CORE
DISRUPTION; HEAT TRANSFER; URANIUM DIOXIDE; STEELS
- 3618 TURBULENT CONVECTION IN HORIZONTAL TWO-
LAYER SYSTEMS WITH AND WITHOUT INTERFACIAL
FREEZING. Bergholz, R.F. (Argonne National
Lab., IL). Trans. Am. Nucl. Soc.; 28: 456-
457 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION;
MELTDOWN; HEAT TRANSFER; HYDRAULICS; TURBULENCE;
CONVECTION
- 3619 EVALUATION OF POSTULATED SODIUM SPILLS
AND FIRES IN AN LMFBR. Tsai, S.S.; Gasser,
R.D.; Albright, D.C. (Brookhaven National
Lab., Upton, NY). Trans. Am. Nucl. Soc.; 28:
458-460 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
CLINCH RIVER BREEDER REACTOR; FIRES; SODIUM;
PIPES; RUPTURES
- 3620 ANALYSIS OF LMFBR CONTAINMENT RESPONSE
TO A CORE-DISRUPTIVE ACCIDENT. Gasser, R.D.;
Pratt, W.T. (Brookhaven National Lab., Upton,
NY). Trans. Am. Nucl. Soc.; 28: 460 (Jun
1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION;
CONTAINMENT BUILDINGS; AFTER-HEAT REMOVAL
- 3621 CORE SUBASSEMBLY CLUSTER DYNAMIC
RESPONSE TO AN ACCIDENTAL PRESSURE RISE. Ash,
J.E. (Argonne National Lab., IL); Kennedy,
J.M.; Belytschko, T.B. Trans. Am. Nucl. Soc.;
28: 462-464 (Jun 1978).
- From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; FUEL ELEMENT CLUSTERS;
PRESSURE GRADIENTS; TRANSIENTS
- 3622 ANALYSIS AND MEASUREMENT OF
SOLIDIFICATION DYNAMICS OF FLOWING TWO-PHASE
NONCONDENSABLE MIXTURES. Greene, G.A.
(Brookhaven National Lab., Upton, NY); Jones,
O.C. Jr.; Kazimi, M.S.; Ginsberg, T.; Barry,
J.J. Trans. Am. Nucl. Soc.; 28: 465-466 (Jun
1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION;
HYDRODYNAMICS; MELTDOWN; SOLIDIFICATION
- 3623 THERMAL CONDITIONS CONTROLLING SUBASSEMBLY
FAILURE PROPAGATION DURING THE TRANSITION PHASE
IN AN LMFBR. Hakim, S.J.; Bauer, T.H.;
McNary, R.D. (Argonne National Lab., IL).
Trans. Am. Nucl. Soc.; 28: 466-468 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION;
HEAT TRANSFER; FUEL ELEMENT CLUSTERS; MELTDOWN
- 3624 TRANSITION PHASE OF FAST REACTOR
OVERPOWER EXCURSIONS. Varela, D.; Kastenber,
W.E.; Dhir, V.K. (Univ. of California, Los
Angeles). Trans. Am. Nucl. Soc.; 28: 468-
469 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; EXCURSIONS; HYDRAULICS;
HEAT TRANSFER; REACTOR CORE DISRUPTION; FLOW
BLOCKAGE
- 3625 ROLE OF RADIATIVE HEAT TRANSFER IN POST-
DISASSEMBLY FUEL SELF-MIXING. Cho, D.H.
(Argonne National Lab., IL); Chan, S.H.
Trans. Am. Nucl. Soc.; 28: 469-470 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION;
HEAT TRANSFER; TEMPERATURE DISTRIBUTION; RADIANT
HEAT TRANSFER
- 3626 EFFECT OF CREEP ON THERMAL FRACTURING IN
OXIDE CRUSTS. Lipinski, R.J. (Sandia Labs.,
Albuquerque, NM). Trans. Am. Nucl. Soc.; 28:
470-471 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION;
MELTDOWN; STEELS; HEAT TRANSFER; BOUNDARY
CONDITIONS; LAYERS; CREEP
- 3627 RECRITICALITY CONSIDERATIONS FOR THE
LOSS-OF-HEAT-SINK ACCIDENT WITH SCRAM. Bari,
R.A.; Ludwig, H.; Pratt, W.T.; Sun, Y.H.
(Brookhaven National Lab., Upton, NY). Trans.
Am. Nucl. Soc.; 28: 471-472 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; CLINCH RIVER BREEDER
REACTOR; MELTDOWN; CRITICALITY
- 3628 FAST INTEGRAL MODEL FOR THE TRANSIENT
ANALYSIS OF A MELTING FUEL PIN. Vaurio, J.K.;

- Mueller, C. (Argonne National Lab., IL).
Trans. Am. Nucl. Soc.; 28: 473-474 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; FUEL ELEMENT FAILURE;
MELTDOWN; LOSS OF FLOW; MATHEMATICAL MODELS
- 3629 SIMPLE COMPUTATIONAL MODEL FOR THE
PREDICTION OF FUEL PIN FAILURE IN THE TRANSIENT
OVERPOWER ACCIDENT. Mast, P.K. (Los Alamos
Scientific Lab., NM); Scott, J.H.; Dornig,
J.J. Trans. Am. Nucl. Soc.; 28: 474-475 (Jun
1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; TRANSIENTS; EXCURSIONS;
FUEL ELEMENT FAILURE; MATHEMATICAL MODELS
- 3630 EFFECT OF POWER SKEW ON THE FTR TOP
ACCIDENT. Yung, S.C.; Wilburn, N.P.
(Westinghouse Hanford Co., Richland, WA).
Trans. Am. Nucl. Soc.; 28: 475-476 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
FFTF REACTOR; EXCURSIONS; TRANSIENTS; POWER
DISTRIBUTION
- 3631 INTERACTION OF MOLTEN UO_2 WITH STAINLESS-
STEEL CLADDING UNDER TOP ACCIDENT CONDITIONS.
Jones, B.G. (Univ. of Illinois, Urbana);
Spencer, B.W.; Henry, R.E. Trans. Am. Nucl.
Soc.; 28: 477-478 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; FUEL-CLADDING
INTERACTIONS; EXCURSIONS; TRANSIENTS; MELTDOWN;
STAINLESS STEELS; URANIUM DIOXIDE
- 3632 BEHAVIOR OF UNIRRADIATED SODIUM-BONDED
CARBIDE DURING MILD OVERPOWER TRANSIENTS.
Tsi, H.C.; Nelmark, L.A. (Argonne National
Lab., IL). Trans. Am. Nucl. Soc.; 28: 478-
479 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; TRANSIENTS; FUEL ELEMENT
FAILURE; URANIUM CARBIDES
- 3633 TRANSIENT FISSION GAS RELEASE EFFECTS ON
CARBIDE FUEL FAILURE DYNAMICS. Henry, T.P.;
Siegmann, E.R. (Combustion Engineering,
Inc., Windsor, CT). Trans. Am. Nucl. Soc.; 28:
479-481 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; FUEL ELEMENT FAILURE;
EXCURSIONS; FISSION PRODUCT RELEASE; URANIUM
CARBIDES; PLUTONIUM CARBIDES
- 3634 TREAT TOP TEST H6: PRELIMINARY RESULTS.
Page, R.J.; Rothman, A.B.; Schmidt, K.J.; Fink,
C.L.; Mouring, R.W.; Lo, R.K.; Froehle, P.H.;
Johanson, E.W. (Argonne National Lab., IL).
Trans. Am. Nucl. Soc.; 28: 481-482 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
FFTF REACTOR; TREAT REACTOR; EXCURSIONS;
HYDRAULICS; MELTDOWN; HYDRODYNAMICS; SIMULATION
- 3635 TREAT TRANSIENT OVERPOWER EXPERIMENT
- R12. Koopman, R.N.; Testa, F.J.; Kramer, N.A.;
McNary, R.O.; Rothman, A.B. (Argonne National
Lab., IL). Trans. Am. Nucl. Soc.; 28: 482-
483 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; EXCURSIONS; TRANSIENTS;
SIMULATION; TREAT REACTOR
- 3636 APPLICABILITY OF TREAT TOP EXPERIMENTS
TO POST-FAILURE DYNAMICS. Padilla, A. Jr.;
Smith, D.E. (Westinghouse Hanford Co.,
Richland, WA). Trans. Am. Nucl. Soc.; 28:
483-484 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; EXCURSIONS; TRANSIENTS;
FUEL ELEMENT FAILURE; HEAT TRANSFER;
HYDRODYNAMICS
- 3637 SENSITIVITY STUDY OF LMFBR CORE
DISASSEMBLIES. Ostensen, R.W. (Sandia
Labs., Albuquerque, NM). Trans. Am. Nucl.
Soc.; 28: 485-486 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION;
EXCURSIONS; MATHEMATICAL MODELS; PRESSURE
GRADIENTS; CONTAINMENT; DAMAGE
- 3638 EFFECTS OF RECENT MODELING DEVELOPMENTS
IN LMFBR PROMPT BURST CALCULATIONS. Sienicki,
J.J.; Abramson, P.B. (Argonne National Lab.,
IL). Trans. Am. Nucl. Soc.; 28: 486-487 (Jun
1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; EXCURSIONS; CRITICALITY;
HYDRODYNAMICS; HEAT TRANSFER
- 3639 PROMPT BURST ENERGETICS EXPERIMENTS:
URANIUM CARBIDE SERIES. Reil, K.O. (Sandia
Labs., Albuquerque, NM); Young, M.F.; Plitz, H.
Trans. Am. Nucl. Soc.; 28: 487-489 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; EXCURSIONS; URANIUM
CARBIDES; HYDRODYNAMICS; THERMODYNAMICS;
SIMULATION
- 3640 FUEL PIN MODELING FOR THE PROMPT BURST
EXCURSION (PBE) EXPERIMENTS. Young, M.F.
(Sandia Labs., Albuquerque, NM). Trans. Am.
Nucl. Soc.; 28: 489-490 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; EXCURSIONS; SIMULATION;
FUEL PINS; FUEL ELEMENT FAILURE
- 3641 COMPARISON OF ICECO PREDICTIONS OF WAVE
PROPAGATION IN THE RIGID TWO-DIMENSIONAL VESSEL
EXPERIMENTS. Wang, C.Y.; Chang, Y.W.
(Argonne National Lab., IL). Trans. Am. Nucl.
Soc.; 28: 490-492 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION;
CONTAINMENT; COMPUTER CALCULATIONS; I CODES;
HYDRODYNAMICS

- 3642 RESPONSE OF FLUID-FILLED VESSELS TO INTERNAL EXPLOSIONS. Bryant, A.R.; Morris, E. (United Kingdom Nuclear Installations Inspectorate, London). Trans. Am. Nucl. Soc.; 28: 492-493(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622---.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION; SIMULATION; PRESSURE GRADIENTS; MATHEMATICAL MODELS; HYDRODYNAMICS
- 3643 APPLICABILITY OF SAREF TO ALTERNATIVE FUELS. Rose, D.; Avery, R.; Dickerman, C.E.; Fauske, H.K.; Mueller, C.J. (Argonne National Labs., IL). Trans. Am. Nucl. Soc.; 28: 496-497(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622---.
SIF REACTOR; REACTOR SAFETY; TEST FACILITIES; EBR-2 REACTOR; THORIUM CYCLE; URANIUM 233
- 3644 FEASIBILITY STUDY FOR A POST-ACCIDENT HEAT REMOVAL FACILITY. Apperson, C.E. Jr.; Barts, E.W.; Bennett, J.G.; Dunwoody, W.E. (Los Alamos Scientific Lab., NM). Trans. Am. Nucl. Soc.; 28: 497-498(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622---.
LMFBR TYPE REACTORS; REACTOR SAFETY; AFTER-HEAT REMOVAL; REACTOR ACCIDENTS; TEST FACILITIES; FEASIBILITY STUDIES
- 3645 EXPERIMENTALLY BASED LIMITS ON SAREF TEST-FUEL-MOTION REACTIVITY FEEDBACK. Mueller, C.J.; Bowers, C.H.; Dickerman, C.E. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 28: 498-499(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622---.
EBR-2 REACTOR; TREAT REACTOR; MELTDOWN; CRITICALITY; REACTOR SAFETY EXPERIMENTS; REACTIVITY
- 3646 FUEL DISPLACEMENT DIAGNOSTICS FOR A 91-PIN SUBASSEMBLY. DeVolpi, A. (Argonne National Lab., IL); Fink, C.L.; Rhodes, E.A.; Hosteny, R.; Rhude, H.V.; Boyar, R.E.; Duncan, L.J.; Evans, A.E.; Barton, J. Trans. Am. Nucl. Soc.; 28: 499-500(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622---.
LMFBR TYPE REACTORS; FUEL ELEMENT CLUSTERS; PARKA REACTOR; MELTDOWN; VOIDS; FUEL ELEMENT FAILURE; HODOSCOPES
- 3647 CODED APERTURE IMAGING IN MANY-PIN FUEL BUNDLES. McArthur, D.A.; Halbleib, J.A. Sr.; Morel, J.E. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 28: 500-501(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622---.
LMFBR TYPE REACTORS; FUEL ELEMENT CLUSTERS; FUEL PINS; IMAGES; MELTDOWN; VOIDS; FUEL ELEMENT FAILURE; NEUTRON DETECTORS; GAMMA DETECTION
- 3648 DESIGN AND OPERATIONAL EVALUATION OF A DIFFERENTIAL PRESSURE LOCAL VOID FRACTION PROBE. Greene, G.A.; Ginsberg, T.; Jones, D.C. Jr. (Brookhaven National Lab., Upton, NY). Trans. Am. Nucl. Soc.; 28: 501-503(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622---.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION; URANIUM DIOXIDE; EVAPORATION; AEROSOLS; SAMPLING
- 3649 CLOSE-IN SAMPLING OF AEROSOLS FROM FISSION-VAPORIZED UO_2 FUEL. Elrick, R.M. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 28: 503-504(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622---.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION; URANIUM DIOXIDE; EVAPORATION; AEROSOLS; SAMPLING
- 3650 FUEL FAILURE PROPAGATION RELIABILITY ASSESSMENT. Brown, R.G. (Westinghouse Advanced Reactors Div., Madison, PA). Trans. Am. Nucl. Soc.; 28: 511-513(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622---.
LMFBR. LMFBR TYPE REACTORS; FUEL ELEMENT FAILURE; PROBABILITY
- 3651 STATISTICAL ANALYSIS OF SIMMER-II RESULTS. Burns, R.D. III; Luck, L.B. (Los Alamos Scientific Lab., NM). Trans. Am. Nucl. Soc.; 28: 513-514(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622---.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION; HEAT TRANSFER; HYDRODYNAMICS; STATISTICS; S CODES; COMPUTER CALCULATIONS; EVALUATION
- 3652 SODIUM INTERACTION WITH LIMESTONE CONCRETE TEST RESULTS. King, D.L.; Sallach, R.A.; Smaardyck, J.E.; Sutherland, H.J. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 28: 521-522(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622---.
LMFBR TYPE REACTORS; MELTDOWN; SODIUM; CONCRETES; CHEMICAL REACTIONS; DECOMPOSITION
- 3653 CHEMICAL PHENOMENOLOGY OF THE SODIUM/LIMESTONE CONCRETE INTERACTION. Sallach, R.A.; Smaardyck, J.E.; King, D.L. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 28: 522-523(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622---.
LMFBR TYPE REACTORS; SODIUM; CONCRETES; CHEMICAL REACTIONS; DECOMPOSITION; TESTING; MELTDOWN
- 3654 MEASUREMENT OF THE PENETRATION OF LIQUID SODIUM INTO LIMESTONE CONCRETE. Sutherland, H.J.; Kent, L.A. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 28: 523-524(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622---.
LMFBR TYPE REACTORS; SODIUM; MELTDOWN; CONCRETES; CHEMICAL REACTIONS
- 3655 INTERACTIONS BETWEEN MOLTEN-CORE DRBRIS AND REACTOR MATERIALS. Swanson, D.G. (Aerospace Corp., Los Angeles); Zehms, E.H.; van Paassen, H.L.; Marchese, A.R. Trans. Am. Nucl. Soc.; 28: 524-525(Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

- (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR MATERIALS; CORIUM;
MELTDOWN; STEELS; URANIUM DIOXIDE; SODIUM;
CONCRETES; CHEMICAL REACTIONS
- 3656 OBSERVATIONS ON EFFECTS OF STAINLESS
STEEL ON UO_2 -GRAPHITE INTERACTIONS.
Heiberger, J.J.; Fink, J.K.; Leibowitz, L.
(Argonne National Lab., IL). Trans. Am. Nucl.
Soc.; 28: 527-528 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; CORE CATCHERS; URANIUM
DIOXIDE; GRAPHITE; STAINLESS STEEL-304; CHEMICAL
REACTIONS; REACTOR MATERIALS; CHEMICAL REACTION
KINETICS
- 3657 LMFBR DESIGN TRANSIENTS AND REGULATORY
REQUIREMENTS. Denise, R.P.; O'Reilly, P.D.;
Speis, T.P.; Holz, H.B.; Riehm, P.F. (Nuclear
Regulatory Commission, Washington, DC).
Trans. Am. Nucl. Soc.; 28: 587-588 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR SAFETY; DESIGN;
REGULATORY GUIDES; LEGAL ASPECTS; ENGINEERED
SAFETY SYSTEMS; REACTOR ACCIDENTS; TRANSIENTS
- 3658 NUMERICAL SIMULATION OF THERMOHYDRAULIC
TRANSIENTS IN LMFBR SYSTEMS. Yang, J.W.;
Madni, I.K.; Agrawal, A.K. (Brookhaven
National Lab., Upton, NY). Trans. Am. Nucl.
Soc.; 28: 588-590 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR SAFETY;
TRANSIENTS; HYDRAULICS; HEAT TRANSFER;
MATHEMATICAL MODELS
- 3659 EXPERIENCE WITH TRANSIENT TESTING AND
SIMULATION AT EBR-II. Stackett, J.I.; Dean,
E.M.; Larson, H.A. (Argonne National Lab.,
Idaho Falls). Trans. Am. Nucl. Soc.; 28: 590-
592 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
EBR-2 REACTOR; TRANSIENTS; REACTOR SAFETY; LOSS
OF FLOW; REACTOR STABILITY; ROD DROP ACCIDENTS;
REACTIVITY INSERTIONS
- 3660 PLANNED NATURAL-CIRCULATION TESTING IN
THE FAST FLUX TEST FACILITY. Bouchev, G.D.
(Dept. of Energy, Washington, DC); Additon,
S.L.; Parziale, E.A. Trans. Am. Nucl. Soc.;
28: 592-594 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
FFTF REACTOR; REACTOR SAFETY; REACTOR
ACCIDENTS; AFTER-HEAT REMOVAL; NATURAL CONVECTION;
HEAT TRANSFER; PERFORMANCE; TEMPERATURE GRADIENTS
- 3661 SIMULATION OF FFTF HOT-LEG RUPTURES AND
LOSS-OF-FLOW EVENTS. Additon, S.L.; Chien,
Y.M. (Westinghouse Hanford Co., Richland,
WA). Trans. Am. Nucl. Soc.; 28: 594-595 (Jun
1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
FFTF REACTOR; REACTOR SAFETY; LOSS OF FLOW;
AFTER-HEAT REMOVAL; TEMPERATURE GRADIENTS;
SIMULATION
- 3662 ALICE: A HYBRID LAGRANGIAN--EULERIAN
CODE FOR CONTAINMENT ANALYSIS. Chu, H.Y.
(Argonne National Lab., IL). Trans. Am. Nucl.
Soc.; 28: 409-411 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION;
CONTAINMENT; COMPUTER CODES; A CODES;
HYDRODYNAMICS
- 3663 SIMPLE MULTICHANNEL THERMAL-HYDRAULIC
MODEL FOR STUDY OF THE TRANSIENT BEHAVIOR OF
COOLANT IN LMFBR SUBASSEMBLY. Chen, W.L.;
Ishii, M.; Grolmes, A.A. (Argonne National
Lab., IL). Trans. Am. Nucl. Soc.; 28: 429-
430 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; TRANSIENTS; FUEL ELEMENT
CLUSTERS; TEMPERATURE DISTRIBUTION; HYDRAULICS;
MATHEMATICAL MODELS; HEAT TRANSFER
- 3664 FUEL PIN COOLABILITY IN LOW POWER
VOIDING. Dunn, F.E. (Argonne National Lab.,
IL). Trans. Am. Nucl. Soc.; 28: 430-431 (Jun
1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; AFTER-HEAT REMOVAL; FUEL
PINS; FUEL ELEMENT FAILURE; VOIDS; MELTDOWN;
REACTOR ACCIDENTS
- 3665 SAS3D PREDICTION OF INTRASUBASSEMBLY
BOILING INCOHERENCE. Dunn, F.E. (Argonne
National Lab., IL). Trans. Am. Nucl. Soc.; 28:
431-432 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; FUEL ELEMENT CLUSTERS;
BOILING; COMPUTER CALCULATIONS; S CODES; HEAT
TRANSFER
- 3666 ESTABLISHMENT OF BUDYANCY-INDUCED
NATURAL CIRCULATION IN LOOP-TYPE LMFBRs.
Khatid-Rahbar, M. (Brookhaven National Lab.,
Upton, NY); Cady, K.B. Trans. Am. Nucl. Soc.;
28: 432-433 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; AFTER-HEAT REMOVAL;
PRIMARY COOLANT CIRCUITS; NATURAL CONVECTION
- 3667 EXPERIMENTAL AND ANALYTICAL STUDY OF
TRANSIENTS IN LMFBR OUTLET PLENUMS. Carbajo,
J.J. (Univ. of Illinois, Urbana); Howard,
P.A. Trans. Am. Nucl. Soc.; 28: 433-434 (Jun
1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; TRANSIENTS; REACTOR
INTERNALS; HYDRAULICS; HEAT TRANSFER
- 3668 HYDRAULIC TESTS ON AN INHERENT SECONDARY
SHUTDOWN ASSEMBLY FOR LMFBRs. Specht, E.R.;
Paschall, R.K.; Chung, D.K.; Jackola, A.S.
(Atomics International, Canoga Park, CA).
Trans. Am. Nucl. Soc.; 28: 435-436 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA
(18 Jun 1978).
See CONF-780622--.

- LMFBR TYPE REACTORS; REACTOR SHUTDOWN; SCRAM; TESTING
- 3669 CREEP AND SWELLING IN EBR-II REACTOR COMPONENTS. Walters, L.C. (Argonne National Lab., Idaho Falls). Trans. Am. Nucl. Soc.; 28: 140-141 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
REACTOR MATERIALS; CREEP; SWELLING; STAINLESS STEELS; PHYSICAL RADIATION EFFECTS; STAINLESS STEEL-316; STAINLESS STEEL-304L; EBR-2 REACTOR
- 3670 IN-REACTOR CREEP MEASUREMENTS. Gilbert, E.R.; Chin, B.A. (Westinghouse Hanford Co., Richland, WA). Trans. Am. Nucl. Soc.; 28: 141-142 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
STAINLESS STEEL-316; STAINLESS STEEL-304; NIMONIC PE16; INCONEL 706; PHYSICAL RADIATION EFFECTS; EBR-2 REACTOR; CREEP; SWELLING
- 3671 FAST-NEUTRON SWELLING OF TYPE 304 STAINLESS STEELS. Flinn, J.E. (Argonne National Lab., Idaho Falls); Weiner, R.A.; Hofman, G.L. Trans. Am. Nucl. Soc.; 28: 224-225 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
STAINLESS STEEL-304; PHYSICAL RADIATION EFFECTS; SWELLING; STAINLESS STEEL-304L; REACTOR MATERIALS; FAST NEUTRONS; EBR-2 REACTOR
- 3672 PERFORMANCE OF TI-MODIFIED TYPE 316 STAINLESS-STEEL CLADDING IN MIXED-OXIDE FUEL PINS. Lawrence, L.A.; Weber, J.W. (Westinghouse Hanford Co., Richland, WA). Trans. Am. Nucl. Soc.; 28: 225-227 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
STAINLESS STEEL-316; TITANIUM ADDITIONS; FUEL CANS; PHYSICAL RADIATION EFFECTS; EBR-2 REACTOR; FAST NEUTRONS
- 3673 LMFBR REFUELING FORCE LIMITS CRITERIA, TESTING, AND EVALUATION. Fox, J.N. (General Electric Co., Sunnyvale, CA). Trans. Am. Nucl. Soc.; 28: 698 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORES; REACTOR FUELING; STRESSES
- 3674 SENSITIVITY ANALYSES OF FAST REACTOR SYSTEMS INCLUDING THORIUM AND URANIUM. Marable, J.H.; Weisbin, C.R. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 28: 718-719 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; FUEL CYCLE; REACTOR KINETICS; URANIUM 233; THORIUM 232; NUCLEAR DATA COLLECTIONS
- 3675 INTEGRAL ALPHA AND POINT CONVERSION RATIO MEASUREMENTS IN ADVANCED FUELS CRITICAL ASSEMBLIES. Bretscher, M.M.; Morman, J.A. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 28: 783-785 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORES; REACTOR KINETICS; PLUTONIUM CARBIDES; URANIUM CARBIDES; MOCKUP; CAPTURE-TO-FISSION RATIO; REACTIVITY WORTHS; COMPARATIVE EVALUATIONS
- 3676 PROCEDURES FOR MEASURING BOWING/ EXPANSION REACTIVITY PARAMETERS IN AN LMFBR. Fox, J.N. (General Electric Co., Sunnyvale, CA). Trans. Am. Nucl. Soc.; 28: 789-790 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; REACTOR CORES; DEFORMATION; REACTIVITY COEFFICIENTS
- 3677 CONTROL ROD INTERACTIONS IN ZPPR-7G, A HETEROGENEOUS LMFBR BENCHMARK ASSEMBLY. Collins, P.J.; McFarlane, H.F.; Carpenter, S.G. (Argonne National Lab., Idaho Falls). Trans. Am. Nucl. Soc.; 28: 782-783 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
LMFBR TYPE REACTORS; CONTROL ELEMENTS; CONTROL ROD WORTHS; REACTOR CORES; CONFIGURATION; MOCKUP
- 3678 FFTF STATUS. Mangelsdorf, T.A. (Westinghouse Hanford Co., Richland, WA). Trans. Am. Nucl. Soc.; 28: 704-705 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
PLANNING; FFTF REACTOR
- 3679 IRRADIATION TEST PROGRAM FOR FFTF. Corrigan, D.C.; Last, G.A. (Westinghouse Hanford Co., Richland, WA). Trans. Am. Nucl. Soc.; 28: 705-707 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
FFTF REACTOR; REACTOR EXPERIMENTAL FACILITIES; IRRADIATION CAPSULES
- 3680 RESPONSE OF EBR-II DELAYED-NEUTRON MONITOR TO TRAMP URANIUM AND A FISSION PRODUCT TEST SOURCE. Fryer, R.M. (Argonne National Lab., Idaho Falls); Reyes, L.A.; Strain, R.V. Trans. Am. Nucl. Soc.; 28: 710-711 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
EBR-2 REACTOR; DELAYED NEUTRONS; RADIATION MONITORS; PERFORMANCE
- 3681 CALCULATION OF RESONANT REACTION RATES NEAR REGION INTERFACES IN THE FTR. Rothrock, R.B. (Westinghouse Hanford Co., Richland, WA). Trans. Am. Nucl. Soc.; 28: 774-776 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
FFTF REACTOR; NEUTRON REFLECTORS; REACTION KINETICS
- 3682 RADIAL-REACTIVITY-COEFFICIENT WEIGHTING SCHEMES FOR A FAST REACTOR. Grimm, K.N. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 28: 785-786 (Jun 1978).
From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).
See CONF-780622--.
EBR-2 REACTOR; REACTOR KINETICS; REACTIVITY

COEFFICIENTS;DISTURBANCES

- 3683 DELAYED-NEUTRON PARAMETERS FOR THE FFTF. Daughtry, J.W.; Dobbin, K.D. (Westinghouse Hanford Co., Richland, WA). Trans. Am. Nucl. Soc.; 28: 790-792 (Jun 1978).

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

See CONF-780622--.

FFTF REACTOR; REACTOR KINETICS; DELAYED NEUTRON PRECURSORS; MATHEMATICAL MODELS

- 3684 EXPERIMENTAL STUDY OF REACTIVITY WORTHS OF SIMULATED LMFBR MELTDOWN CONFIGURATIONS. Bhattacharyya, S.K.; Smith, D.M.; McKnight, R.D.; LeSage, L.G. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 28: 786-788 (Jun 1978).

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

See CONF-780622--.

LMFBR TYPE REACTORS; MELTDOWN; REACTOR SAFETY; REACTIVITY WORTHS; MATHEMATICAL MODELS

- 3685 ENDF/B-V CONTENT AND PERFORMANCE IN FAST REACTOR SYSTEMS. McKnight, R.D. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 28: 735-736 (Jun 1978).

From ANS annual meeting; San Diego, CA, USA (18 Jun 1978).

See CONF-780622--.

NUCLEAR DATA COLLECTIONS; THORIUM 232 TARGET; URANIUM 233 TARGET; URANIUM 238 TARGET; PLUTONIUM 239 TARGET; NEUTRON REACTIONS; CROSS SECTIONS; FAST REACTORS; LMFBR TYPE REACTORS; ZPR-6 REACTOR; REACTOR KINETICS

- 3686 EVALUATION OF COMPUTATIONAL TECHNIQUES FOR LMFBR SAFETY ANALYSIS. Herrmann, W. (Sandia Labs., Albuquerque, N.Mex. (USA)); Gross, M. (Science Applications, Inc., Oakland, Calif. (USA)). Nucl. Eng. Des.; 48: No. 1, 81-95 (Jun 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, USA (15 - 19 Aug 1977).

Analysis of Primary Containment Transients (APRICOT) is an ERDA sponsored project in which a variety of reactor safety analysis groups around the world have been invited to participate by performing calculations to verify capabilities of large computer codes used to analyze postulated core disruptive accidents of liquid metal fast breeder reactors. Nine groups have performed calculations of the first three problems which were set, using ten computer codes. Two problems were simple test problems for which analytical solutions exist, namely an ideal gas shock tube, and a suddenly pressurized spherical cavity in an infinite elastic medium. The third problem concerns an explosion in a partially water-filled over-strong cylindrical containment vessel for which experimental data exist. A critique of the results of these calculations is given in this paper.

- 3687 ALTERNATIVE NUCLEAR FUEL CYCLES: THE IMPLICATIONS ON URANIUM RESOURCE UTILIZATION. Till, C.E.; Chang, Y.I. (Argonne National Lab., IL). Combustion; 49: No. 12, 22-26 (Jun 1978).

The authors attempt to summarize the implications of fuel-cycle selection by criteria based on proliferation grounds and on achievable nuclear electric capacities within a limited natural resource base. The question is how far can uranium supplies be stretched if criteria of this kind are used to define

admissible reactor cycles. The question is a simple one. The problem in answering it lies in the very number of reactor types, fuel cycles, and deployment permutations; and the widely varying opinion currently held on both the magnitude of the uranium resource base and the amount and time span of nuclear power that may represent an adequate objective. 1 ref.

- 3688 (JAPFNR--317) FAST BREEDER REACTOR DEVELOPMENT IN JAPAN. PROGRESS REPORT, JULY--SEPTEMBER 1977. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Jun 1978. Translation of PNC-N--251-78-01 17p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

Progress in LMFBR development in Japan is summarized. The development program has progressed from the experimental JOYO Reactor to design work and mockup testing of components for the MONJU prototype reactor.

- 3689 ULTRASONIC INSPECTION METHODS FOR SMALL-BORE APPLICATIONS. Cook, K.V.; Cunningham, R.A. Jr. Mater. Eval.; 36: No. 7, 49-54 (Jun 1978).

The nondestructive inspection of tubing and tube-to-tubesheet weldments in heat exchangers for reactor systems is necessary to ensure the high integrity essential for safe, reliable, continuously operating units. An ultrasonic inspection system has been developed for small-bore applications. The current system can be used to inspect tubing or weldments with bores as small as 10.1 mm (0.4 in.) for transversely or longitudinally oriented discontinuities; however, similar inspections are possible within even smaller bores. Since the probe head is rotated to provide inspection along a helical path, special ultrasonic signal connectors are employed. The miniature probe heads employed are designed for rapid exchange in the mechanical scanner. These probes use commercially available transducers and are capable of performing various ultrasonic inspections with both single and dual examination techniques. The inspection methods and equipment are discussed, along with results of calibration, recording, and correlation studies performed on laboratory-type specimens. Reference notches with depths representing 2 percent of the wall thickness were detected and recorded from the bore of 15.88 mm OD x 2.77 mm wall (0.625 x 0.109 in.) tubing.

- 3690 EXPERIMENTAL INVESTIGATION ON AIR-DISTRIBUTION IN WATER FLOWING THROUGH A 61-ROD BUNDLE WITH HELICAL SPACERS. Chung, M.K. J. Korean Nucl. Soc.; 10: No. 2, 79-86 (Jun 1978). (In English and Korean).

Data were obtained on air-distributions in two-phase up flow in a vertical rod-bundle test-section. The test-section was a hexagonal shaped 61-rod bundle where each rod was wrapped with helical spacers. The variables were flow rates of air and water and air inlet positions. Experimental data were obtained at the outlet of the test-section. Experiments were performed in two parts. First data were taken at increasing flow rates of air keeping water flow rates constant, and second at simultaneous increase of air and water flow rates. At each flow condition, air supply position could be changed to 4 different positions. Data obtained by electrical void-needle technique were analyzed and are presented here in graphical forms for comparison. Results demonstrate qualitatively that air-distribution tends to be more uniform as water flow rates are increased. The air supply positions have noticeable

effects on the pattern of air-distribution.

enables the licensing of fast breeders.

- 3691 EUROPIA AS A NUCLEAR CONTROL MATERIAL. Wu, P.C.S. (Westinghouse Advanced Reactors Div., Madison, PA). Nucl. Technol.; 39: No. 1, 84-94(Jun 1978).

The Physical and nuclear properties of Eu_2O_3 were reviewed, evaluated, and compared with those of the other potential control materials for breeder reactors. Europia with a 90% theoretical density has approximately the same amount of europium atoms per unit volume as the pure metal. Consequently, the reactivity worth per unit volume of Eu_2O_3 is similar to that of pure metal. In addition, the reactivity of Eu_2O_3 is superior to tantalum and is comparable to natural B_4C . The decay heat of Eu_2O_3 is much lower than that of tantalum over the range of decay times that is of interest with respect to handling. Irradiation-induced swelling of Eu_2O_3 is lower than that of B_4C , and postirradiation examination revealed only approx. 1% dimensional changes in the thermal- and/or fast-neutron spectrum ($8 \times 10^{25} \text{ n/m}^2$). Although surface reaction between Eu_2O_3 and the fast test reactor reference cladding Type 316 stainless steel leads to the formation of a europium silicate, it is expected that Eu_2O_3 would be compatible with low-silicon Type 316 stainless steel. In addition, the Eu_2O_3 -sodium system was also shown to be compatible under stimulated liquid-metal fast breeder reactor conditions without significant change. Consequently, Eu_2O_3 is an excellent alternate candidate compared to B_4C as the neutron absorber for fast breeder reactors.

- 3692 COMPOSITION OF THE CORROSION PRODUCT OXIDE PHASES IN THE GAPS OF THREE LMFBR-TYPE MIXED OXIDE FUEL PINS. Walker, C.T. (Commission of the European Communities, Karlsruhe (Germany, F.R.). European Inst. for Transuranium Elements). J. Nucl. Mater.; 74: No. 2, 358-362(Jun 1978).

Identification of the corrosion product oxide phases in the gaps of mixed oxide fuel pins clad with austenitic stainless steel is an important step in understanding the mechanisms of corrosion. Once the oxide is identified the reaction paths by which it is formed can be established. The corrosion product oxide will also be a reliable indication of the oxygen potential in the gap. A grey oxide phase which is recognised to contain the products of cladding corrosion is often observed in the gaps of mixed oxide fuel pins. Its main constituents are known to be chromium, caesium and oxygen, and recently the composition of the grey oxide phase in the gaps of three fuel pins was shown to be broadly similar, and it was proposed that the phase was essentially chromium oxide.

- 3693 CONSTITUTIONALITY OF SECTION 7 OF THE ATOMIC ENERGY ACT: SECTION 20 GG 'KALKAR DECISION'. Z. Energiewirtsch.; No. 2, 156-160(Jun 1978). (In German).

OVG Muenster, decision dated Aug. 18th, 1977 - VII A 338/74: 'Section 7 of the Atomic Energy Act disagrees with the constitution in as far as it also allows the licensing of FBR type reactors'. The grounds upon which the judgment is based are given in detail: According to the opinion of the Senate, section 7 of the Atomic Energy Act does not conform to the principle of separation of powers (section 20, sub-section 2, sentence 2 GG), to the principle of parliamentary democracy (section 20, sub-section 1 and 2 GG) and to the principles of the law and order state (section 20, sub-section 3 GG) in as far as the present version

- 3694 CONSTRUCTION OF THE WORLD'S FIRST FULL-SCALE FAST BREEDER REACTOR. 1. ORGANIZATION OF AN INTERNATIONAL PROJECT. Nucl. Eng. Int.; 23: No. 272, 43-60(Jun 1978).

The 1200 MWe nuclear power station under construction at Creys-Malville in S.W. France contains the fast breeder reactor Super Phenix. A survey is presented which comprises six sections describing the following:- organisation; lay-out and main design; main safety features; design and development of main components; core structure and assemblies; site assembly and construction progress. A coloured cut-away pull-out wall-chart of the reactor and reactor buildings and a list of all main parameters are included.

- 3695 RADIOACTIVITY IN THE PRIMARY CIRCUIT OF THE FAST BOR-60 REACTOR. Kizin, V.D.; Lisitsyn, E.S.; Polyakov, V.I.; Chechetkin, Yu.V. At. Energ. (USSR); 44: No. 6, 492-498(Jun 1978). (In Russian).

The experimental results of the accumulation and distribution study of radioactive corrosion and fission products in the primary circuit of the BOR-60 are presented. It is shown that after the reactor prolonged operation with defective fuel elements, the radiation conditions at repair works are determined by the fission radioactive products. The most radiation-dangerous nuclides are ^{137}Cs , ^{140}Ba , ^{95}Nb . The BOR-60 cold trap purifies the coolant practically completely from iodine radionuclides by 20-90% from cesium radionuclides, by 20-30% from antimony and cobalt. At reactor operating approximately with 1% of defective fuel elements, the dose rate in the primary circuit boxes after sodium draining from the reactor loops equals $70-100 \mu\text{Rxc}^{-1}$, the radiation conditions at the NPP BOR-60 for the whole operation period remaining satisfactory. NPP radioactive releases of ^{41}Ar do not exceed 500 Ci a year.

- 3696 IRRADIATION EFFECTS OF BORON CARBIDE USED AS CONTROL ROD ELEMENTS IN FAST BREEDER REACTORS. Inoue, T.; Onchi, T.; Koyama, H. (Central Research Inst. of Electric Power Industry, Tokyo (Japan)). J. Nucl. Mater.; 74: No. 1, 114-122(Jun 1978).

The paper describes radiation effects on B_4C pellets used as control rod elements in the Enrico Fermi Fast Breeder Reactor. Pellet swelling ($\Delta V/V$) caused by irradiation was less than 1% in which crystal lattice swelling was less than 20%. Many microcracks, a main cause of pellet swelling, appeared in the irradiated pellets. The production of microcracks was related to graphite precipitation in the pellets before irradiation. Open pores which did not exist in the unirradiated pellets were formed in the irradiated ones. In a unit cell of B_4C , the a-axis elongated by 0.025 Å and the c-axis shrank by 0.07 Å by irradiation. Moreover, three recovery stages were found which were from room temperature to 400°C, from 400 to 750°C and from 850 to 1100°C. The recovery mechanisms in the irradiated pellets are discussed in terms of the helium behaviour.

- 3697 COUPLED PROBLEMS IN TRANSIENT FLUID AND STRUCTURAL DYNAMICS IN NUCLEAR ENGINEERING. PT.2. A DISCRETE MODEL FOR SQUEEZE FLOW BETWEEN TWO ADJACENT LMFBR SUBASSEMBLIES. Zehlein, H. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). Appl. Math. Model.; 2: No. 2, 90-98(Jun 1978).

The paper deals with a detailed

investigation of the interaction between sub-assemblies and coolant dynamics within an LMFBR core under transient pressure loading. The one-dimensional discrete model approximates the axial squeeze flow by a combined Lagrangian particle Euler cell approach. The coupling between the fluid and the structure is achieved by a step by step scheme. An energy balance criterion is used to improve its computational efficiency. An extension for two-dimensional squeeze flow, where axial and transverse flow around the subassembly may occur, completes the mathematical model.

- 3698 OPEN ROAD TO CONTROL OF CONSTITUTIONALITY OF SECTION 7 OF THE ATOMIC ENERGY ACT. COMMENTS ON THE INTERIM NOTICE OF THE FEDERAL CONSTITUTIONAL COURT IN THE MATTER OF THE 'FAST BREEDER' AT KALKAR. Rengeling, H.W. (Bonn Univ. (Germany, F.R.)). *Energiewirtschaftl. Tagesfragen*; 28: No. 6, 372-375 (Jun 1978). (In German).

The Federal Constitutional Court (BVerfG), in answer to the court order of the Higher Administrative Court at Muenster, has affirmed in its interim decision of Jan. 31st, 1978 the permissibility of the concrete judicial review of the constitutionality of section 7 as being correct, and thus it has accepted at the same time the responsibility for the decision as being the responsibility of the Federal Constitutional Court. For practical reasons the decision is to be welcomed because it is very likely that the question of the constitutionality of section 7 of the Atomic Energy Act as far as it permits the licensing of FRBs, which has come into the open now on account of the Muenster decision, would have been referred again in the near future to the BVerfG, if the court had considered Muenster's order as not permissible. From the judicial point of view, the BVerfG's grounds are of interest; the permissibility of the judicial review of the constitutionality within the framework of the Kalkar case already having been very much in dispute before the verdict. The arguments of the BVerfG, which could be of great interest for similar proceedings in the future are briefly looked at in a critical manner, because the decision made in Karlsruhe will not remain without impacts on law concerning energy and environmental protection and on relevant policies.

- 3699 COUPLED PROBLEMS IN TRANSIENT FLUID AND STRUCTURAL DYNAMICS IN NUCLEAR ENGINEERING. PT.1. SAFETY PROBLEMS SOLVED BY FLOW SINGULARITY METHODS. Krieg, R. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). *Appl. Math. Model.*; 2: No. 2, 81-89 (Jun 1978).

Some important problems in coupled fluid-structural dynamics which occur in safety investigations of liquid metal fast breeder reactors (LMFBR), light water reactors, and nuclear reprocessing plants are discussed and a classification of solution methods is introduced. A distinction is made between the step by step solution procedure, where available computer codes in fluid and structural dynamics are coupled, and advanced simultaneous solution methods, where the coupling is carried out at the level of the fundamental equations. Results presented include the transient deformation of a two-row pin bundle surrounded by an infinite fluid field, vapour explosions in a fluid container and containment distortions due to bubble collapse in the pressure suppression system of a boiling water reactor. A recently developed simultaneous solution method is presented in detail. Here the fluid dynamics (inviscid,

incompressible fluid) is described by a singularity method which reduces the three-dimensional fluid dynamics problems to a two-dimensional formulation. In this way the three-dynamics fluid dynamics as well as the structural (shell) dynamics can be described essentially by common unknowns at the fluid-structural interface. The resulting equations for the coupled fluid-structural dynamics are analogous to the equations of motion of the structural dynamics alone.

- 3700 SWELLING ANALYSIS OF HIGHLY-RATED MX-TYPE LMFBR FUELS. II. MICROSCOPIC SWELLING BEHAVIOUR. Ronchi, C.; Ray, I.L.F.; Thiele, H.; Laar, J. van de (Commission of the European Communities, Karlsruhe (Germany, F.R.)). *European Inst. for Transuranium Elements*. *J. Nucl. Mater.*; 74: No. 2, 193-211 (Jun 1978).

Microscopic swelling has been investigated by electron microscopy in several MX-type fuels, irradiated in fast and thermal flux. The results show that fission gas bubbles in these compounds grow to large sizes if the in-pile temperature rises above a critical value (swelling critical temperature $T_{sub}(c)$). A comparison has been made of the swelling rates in fuels of different composition, showing that $T_{sub}(c)$ increases from carbides to nitrides. In fuels subjected to in-pile restructuring (highly rated He-bonded pins) microscopic swelling is affected by pore and grain boundary migration. The influence of these phenomena on the fuel swelling performance has been discussed.

- 3701 TOPICAL PROBLEMS OF CORROSION RESEARCH FOR NUCLEAR POWER PURPOSES. Eremias, B. (Statni Vyzkumny Ustav Ochranu Materialu G.V. Akimova, Prague (Czechoslovakia)). *Jad. Energ.*; 24: No. 6, 201-205 (Jun 1978). (In Czech).

Currently, research is focused on stress corrosion, intergranular corrosion, corrosion in water and steam, hydrogen-induced corrosion and corrosion in liquid sodium. The effort to limit stress corrosion resulted in the application of high nickel content austenitic steels. In these steels, the susceptibility to stress corrosion is mainly affected by previous heat treatment and the presence of chloride ions. Attention is also paid to medium and high-alloy chromium steels and susceptibility is studied to intergranular corrosion and stress corrosion. Of low-alloy steels the 21/4Cr-1Mo type steels stabilized with Nb or nonstabilized are studied with respect to decarburization kinetics and changes in mechanical properties in the presence of hydrogen. Of nonferrous metals zirconium alloys are studied used as cladding materials for fuel elements, mainly Zircaloy 2 and 4, with regard to their resistance to high-temperature oxidation, high-pressure steam action, etc.

- 3702 SODIUM AEROSOL GENERATION. Allen, M.D.; Briant, J.K.; Kaune, W.T.; Craig, D.K. (Battelle Pacific Northwest Labs., Richland, WA (USA)). *Health Phys.*; 34: No. 6, 39-54 (Jun 1978).

A method for generating a respirable sodium aerosol at a constant rate is described for use in rodent inhalation studies. These studies will assess the toxicologic effect of aerosols which may be formed if pipes carrying liquid sodium in a LMFBR rupture. The mass generation rate and aerodynamic size distribution of the aerosol are characterized.

- 3703 FLOW DUCT FOR NUCLEAR REACTORS.

Straalsund, J.L. (to Dept. of Energy). US Patent 4,089,743. 16 May 1978. Filed date 21 Jan 1977. 6p.

PAT-APPL-761,038.

Improved liquid sodium flow ducts for nuclear reactors are described wherein the improvement comprises varying the wall thickness of each of the walls of a polygonal tubular duct structure so that each of the walls is of reduced cross-section along the longitudinal center line and of a greater cross-section along wall junctions with the other walls to form the polygonal tubular configuration.

3704 NUCLEAR REACTOR FUEL ELEMENT ASSEMBLIES. Betts, C.; Collinson, A.E.; Hart, D.F. British Patent 1,510,127/A/. 10 May 1978. 7p.

Reactor fuel element assemblies are described having axial flow cavitation suppressing impedances. The latter comprise a tubular member housing a series of flow defining plates arranged transversely to the longitudinal axis of the tubular member, the series being formed from a first and a second set of plates clamped into face-to-face abutment and arranged so that a plate of one set alternates, as viewed in the direction of flow, with a plate of the other set. The first set of plates is of woven wire mesh construction and the second set have aligned cylindrical apertures that define the present impedance flow area. The arrangement is particularly suitable for liquid metal cooled fast breeder reactors.

3705 APPARATUS AND METHOD FOR THERMAL POWER GENERATION. Cohen, P.; Redding, A.H. (to Westinghouse Electric Corp.). US Patent 4,087,985. 9 May 1978. Filed date 17 Mar 1976. 8p.

An improved thermal power plant and method of power generation is described which minimizes thermal stress and chemical impurity buildup in the vaporizing component, particularly beneficial under loss of normal feed fluid and startup conditions. The invention is particularly applicable to a liquid metal fast breeder reactor plant.

3706 RADIONUCLIDE TRAP. McGuire, J.C. (to Dept. of Energy). US Patent 4,088,533. 9 May 1978. Filed date 18 Jan 1977. 8p.

PAT-APPL-760,301.

The deposition of radionuclides manganese-54, cobalt-58 and cobalt-60 from liquid sodium coolant is controlled by providing surfaces of nickel or high nickel alloys to extract the radionuclides from the liquid sodium, and by providing surfaces of tungsten, molybdenum or tantalum to prevent or retard radionuclide deposition.

3707 TEMPERATURE CONTROL SYSTEM FOR A J-MODULE HEAT EXCHANGER. Basdekas, D.L.; Macrae, G.; Walsh, J.M. (to Dept. of Energy). US Patent 4,088,182. 9 May 1978. Filed date 29 May 1974. 4p.

PAT-APPL-474,287.

The level of primary fluid is controlled to change the effective heat transfer area of a heat exchanger utilized in a liquid metal nuclear power plant to eliminate the need for liquid metal control valves to regulate the flow of primary fluid and the temperature of the effluent secondary fluid.

3708 NUCLEAR REACTOR. Chevallier, R.;

Sauvage, M. (to Societe Generale de Constructions Electriques et Mecaniques Alsthom S.A.). US Patent 4,087,325. 2 May 1978. Priority date 29 Dec 1975, France, 6p.

Described is an integrated nuclear reactor cooled by a liquid metal and constituted by a main tank, which contains a central inner tank containing the core and the hot liquid metal, this inner tank being connected to individual cylindrical chambers each containing a heat exchanger wherein the surfaces of the inner tank and of the cylindrical chambers bulge radially outwards at the level of supply orifices for transferring the hot liquid metal from the inner tank to the heat exchangers and wherein the cylindrical chambers are fixed directly onto the inner tank by neck portions interconnecting the two bulges and providing a path for liquid metal flow.

3709 EFFECT OF FINITE PLANT LIFE ON THE DOUBLING TIME OF A BREEDER REACTOR POPULATION. Lee, S.M. (Reactor Research Center, Tamilnadu, India). Nucl. Technol.; 38: No. 3, 384-386 (May 1978).

When very large values are obtained for the doubling time of a breeder reactor type, it becomes interesting to see how the finiteness of the lifetime of a reactor plant affects the growth of a breeder population. A possible effect is the time lag between expiration of one plant and start of operation of its replacement plant. An analytical expression for the necessary correction to the compound system doubling time of the reactor population can be derived. Numerical examples show that in realistic cases, correction to the compound system doubling time will be small.

3710 CONDENSED REVIEW OF THE TECHNOLOGY OF POST-ACCIDENT HEAT REMOVAL FOR THE LIQUID-METAL FAST BREEDER REACTOR. Kazimi, M.S. (Massachusetts Inst. of Tech., Cambridge); Chen, J.C. Nucl. Technol.; 38: No. 3, 339-366 (May 1978).

The analyses performed to evaluate the capacity to contain the core materials after a hypothetical core meltdown accident are generally grouped under the area of post-accident heat removal (PAHR). Substantial advancement in the state-of-the-art of the PAHR area has occurred within the last few years. The key issues of PAHR, as applied to safety analyses of liquid-metal breeder reactors, are summarized. The review includes heat source strength for post-accident conditions, in-vessel core debris retention, ex-vessel core debris retention, and the effects of alternate fuel materials.

3711 (BNWL-tr--326) PRINCIPLES OF DETERMINING DAMAGE TO THE FUEL ELEMENTS OF FAST POWER REACTORS BY MONITORING SODIUM AND GAS ACTIVITY. Iefimov, I.A.; Kulakovskii, M.Ya.; Filonov, V.S. May 1978. Translated from pp 399-416 of engineering of fast reactors for safe and reliable operation (international conference) proceedings, Vol. I, Karlsruhe, Germany, October 9-13, 1972 (CONF-721006--13). 20p. Dep. NTIS, PC A02/MF A01.

From International conference on engineering of fast reactors for safe and reliable operation; Karlsruhe, F.R. Germany (9 Oct 1972).

This paper states the main requirements which have to be met by systems for determining damage to the fuel elements of fast power reactors by delayed neutrons, gas activity, and sodium coolant activity monitoring. The influence of reactor layout, detector geometry, neutron background, and gamma background on the

sensitivity of such systems is examined. The paper contains theoretical data on the magnitude of the effective signal resulting from damage to fuel elements in the BN-350 and BN-600 reactors and experimental data relating to the use of such systems in determining fuel element damage during operation in the BR-5 reactor.

- 3712 WATER LEAKS IN SODIUM-HEATED FAST REACTOR BOILERS. Hayes, D.J. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Phys. Technol.; 9: No. 3, 96-100(May 1978).

Constraints on plant design which may result from considerations of leak behaviour and leak detection limits are briefly considered. The sodium-water interface and reactions, the behaviour of small leaks, hydrogen bubbles and detection methods, including galvanic cell methods, are included.

- 3713 CORE DESIGN: FUEL MANAGEMENT AS A BALANCE BETWEEN IN-CORE EFFICIENCY AND ECONOMICS. Robichaud, G.G.; Tulenko, J.S.; Strauss, S.D. (Babcock and Wilcox Co). Power (N.Y.); 122: No. 5, 52-57(May 1978).

After a brief review of the steps involved in the transformation of enriched uranium into the highly engineered fuel assemblies that drive a nuclear reactor, a state-of-the-art survey of fuel-management techniques is given. To optimize the nuclear fuel cycle, the industry is using available fuel-management techniques, such as multibatch shuffling, use of burnable poisons, coast-down, and fuel-assembly reinsertion. In addition, new techniques are under investigation, including designing for a wetter lattice (higher ratio of water to fuel in the core), use of gadolinium as a burnable poison in pressurized-water reactors (PWRs), increasing fuel burnup limits, and designing for minimum use of poison structural material within the core.

- 3714 INFLUENCE OF SODIUM ENVIRONMENT ON THE UNIAXIAL TENSILE BEHAVIOR OF TITANIUM MODIFIED TYPE 316 STAINLESS STEEL. Natesan, K.; Chopra, O.K.; Kassner, T.F. (Argonne Natl Lab, Ill). J. Nucl. Mater.; 73: No. 2, 137-150(May 1978).

True stress-true strain tensile data have been obtained for titanium-modified type 316 stainless steel in the solution-annealed condition and after exposure to a flowing sodium environment at temperature of 700, 650, 600 and 550°C. The specimens were exposed between 120 and 5012 h to produce carbon penetration depths in the range 0.05-0.30 mm. When compared with solution annealed specimens, the tensile flow behavior of the sodium-exposed specimens is characterized by a higher strain hardening rate, which decreases rapidly as the flow stress increases. The loss in tensile ductility of the material due to carburization in sodium environment was minimal.

- 3715 SIMULATION EXPERIMENTS ON LOCAL COOLING CHANNEL BLOCKAGES IN TYPICAL SNR FUEL ELEMENT CLUSTERS. Basmer, P.; Schultheiss, G.F. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). Atomwirtsch., Atomtech.; 23: No. 5, 218-220(May 1978). (In German).

TL

- 3716 DRIED HOT CONCRETE VESSEL FOR NUCLEAR REACTORS: PROPOSAL OF A NEW DESIGN CONCEPT. Bazant, Z.P. (Northwestern Univ., Evanston,

Ill. (USA). Dept. of Civil Engineering); Fistedis, S.H. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 47: No. 2, 317-324(May 1978).

The paper discusses special features of the concept of a dried hot prestressed concrete reactor vessel for pool-type liquid metal cooled fast breeder reactors. In particular, the potentially advantageous feature of removing evaporable water from concrete when it is being kept hot is discussed along with its technological implications.

- 3717 ASSESSMENT OF FUEL FREEZING AND DRAINAGE PHENOMENA IN A REACTOR SHIELD PLUG FOLLOWING A CORE DISRUPTIVE ACCIDENT. El-Genk, M.; Cronenberg, A.W. (New Mexico Univ., Albuquerque (USA). Dept. of Nuclear Engineering). Nucl. Eng. Des.; 47: No. 2, 195-225(May 1978).

An important problem related to the assessment of the recriticality potential for an LMFBR following a core disruptive accident is an understanding of the freezing phenomena of molten fuel on a cold structure which may prevent fuel dispersal and subsequent shutdown. Transient analytical freezing and drainage calculations have been applied to molten UO_2 travel through the rather cold lower shield plug of the Clinch River Breeder Reactor (CRBR). The successive approximation technique is used to obtain a solution of the non-linear freezing problem, where such effects as heat generation, viscous heat dissipation, temperature dependent thermophysical properties and a convective boundary condition at the solidification front have been incorporated into the present analytical formulation. Results indicate that previous steady-state analyses overestimated the rate of frozen layer build-up by about a factor of two. However, of primary importance is the driving force for drainage and the diameter of the shield plug flow channel.

- 3718 VISCOSITY OF MOLTEN ALUMINA. Blomquist, R.A.; Fink, J.K.; Leibowitz, L. (Argonne National Lab., IL). Am. Ceram. Soc. Bull.; 57: No. 5, 522(May 1978).

Viscosity data on molten ceramics are needed in the analysis of core containment and heat-removal associated with hypothetical core disruptive accidents in LMFBR reactors. The oscillating-cup technique was used to determine the viscosity of alumina. (FS)

- 3719 LONG-TERM OPTIONS FOR THE FR FUEL CYCLE. Flowers, R.H.; Johnson, K.D.B.; Miles, J.H.; Webster, R.K. (UKAEA Research Group, Harwell. Atomic Energy Research Establishment). Atom (London); No. 259, 138-142(May 1978).

From 5. Energy technology conference; Washington, USA (Feb 1978).

This paper is designed to complement and amplify the general philosophy developed by Dr. W. Marshall (UKAEA) in his Graham Young Memorial Lecture "Nuclear power and the proliferation issue", February 1978. The position at say about the year 2050 is discussed, taking the Liquid Metal Fast Breeder Reactor as the typical reactor. It is assumed that by then fast reactors will have completely displaced thermal reactors from the power system. The safeguarding of Pu is considered. Headings are as follows -Reasons for reprocessing in a fast reactor fuel cycle; fuel processing procedures in a fully evolved LMFBR cycle (processing requirements set by the fast reactor system; possible developments in LMFBR fuel processing; co-processing of U with Pu; fuel fabrication by the gel process; integrated fuel plants); potential further developments

for reducing plutonium accessibility; relaxation of specifications on fission product levels in new fast reactor fuel; and enhanced recycle of fission products.

- 3720 CHOICE OF SYSTEMS AND OPTIMUM USE OF NUCLEAR MATERIAL RESOURCES. Ferrari, A.; Kresmer, J.; Pierre, C. (CEA). Ann. Mines; 184: No. 5-6, 77-96(May-Jun 1978). (In French).

The authors review the salient features of key nuclear systems and give, for each of these, the worldwide distribution of existing power reactors, and those under construction or contracted for. The review covers the Light Water Reactors (LWR) which account for 77% of all electric power generated by the former group of reactors and for 89% by the latter, with a clear trend toward the Pressurized Water Reactors (PWR). Some advanced systems such as the High Temperature Reactors (HTR) and the Liquid Metal Fast Breeder Reactors (LMFBR) are also discussed. Factors that must be considered when selecting a nuclear system are examined, and an explanation is given of how the French chose the PWR system complemented by breeder reactors.

- 3721 NUCLEAR POWER IN THE SOVIET UNION. Nucl. Can./Can. Nucl.; 17: No. 5, 6-7(May 1978). (In English and French).

Thermal reactors in the Soviet Union are of two kinds, the PWR and the graphite moderated water cooled boiling water reactor (LGR). During the period 1971-6, 85 billion kWh of nuclear electricity were produced. A 350 MWe LMFBR near Shevchenko achieved criticality in 1973. For PWR reactors, 440 and 1000 MWe are standard sizes; for LGR's, 1 GWe. It is intended to produce a series of 1.6 GWe LMFBR's. The Soviet export agency has installed or is constructing nuclear stations in Bulgaria, Finland, the German Democratic Republic, Hungary, Poland and Romania. F.R. Germany has used Soviet enrichment capacity.

- 3722 SPATIAL MESH EFFECT IN THREE-DIMENSIONAL DIFFUSION CALCULATIONS FOR FAST REACTORS. Sakagami, M.; Takeda, S. (Hitachi Ltd., Ibaraki (Japan). Energy Research Lab.). J. Nucl. Sci. Technol. (Tokyo); 15: No. 5, 365-375(May 1978).

A correction to the numerical solution of the diffusion equation solved using the finite mesh spacing method has been studied for a fast breeder reactor. It has been found that the correction to the effective multiplication factor $k_{\text{sub}}(\text{eff})$ calculated with the finite mesh spacing method can be expressed approximately as $-\Delta k(\Delta r, \Delta z) = k - k_{\text{sub}}(D)(\Delta r, \Delta z)$ asymptotically equals $-[(\Delta r)^2(c_1\rho + c_2) + (\Delta z)^2(c_3\rho + c_4)]$, where k and $k_{\text{sub}}(D)(\Delta r, \Delta z)$ are, respectively, the eigenvalues of the diffusion equation and its approximate finite difference equation with mesh spacings Δr in the x-y plane and Δz along the z axis. The quantity ρ is the total reactivity worth of control rods inserted into the core. The constants c_1 , c_2 , c_3 and c_4 are determined from the geometrical shape and atomic densities of constituent materials of a reactor core. For a prototype LMFBR, Δk with $\Delta r = \Delta z = 11.5$ cm (hexagonal-z mesh model) amounts to 0.8% Δk for the state with all control rods fully withdrawn or 1.9% Δk for the state with control rods fully inserted. The Δk 's with $\Delta r = \Delta z = 3.83$ cm (triangular-z mesh model) are less than 0.23% Δk for various patterns of the control rods. As for the power distribution, the hexagonal-z mesh model overestimates the power near the mid plane of

the core with all control rods fully withdrawn (inserted), by 0 -- 1.6% (0 -- 3.9%) in the inner core and by 0 -- 3.7% (0 -- 4.8%) in the radial blanket, but underestimates it by 0 -- 1.4% (0 -- 1.6%) in the outer core, compared with that from the triangular mesh model.

- 3723 ACCIDENT ALARM IN STEAM GENERATORS IN SODIUM COOLED FAST REACTOR POWER PLANTS. III. EXPERIMENTAL RESEARCH IN FLOW FLUCTUATIONS IN PIPES FOLLOWING WATER LEAK IN SODIUM. Banovec, J.; Matal, O. (Vyzkumny Ustav Zavodu Energetického Strojirenstvi, Brno-Slatina (Czechoslovakia)). Jad. Energ.; 24: No. 5, 162-166(May 1978). (In Czech).

The fluctuations of sodium flow rate due to water injected into the sodium were measured on an experimental sodium loop using two electromagnetic flowmeters. A leak of water into sodium of 0.01 g/s was reliably detected above the level of the noise signal of the flowmeter. The ratio of the signal of the flowmeter that was placed at a greater distance from the point of leak to the signal of the flowmeter nearer to the point of leak changed with the flow rate of sodium and was 2 to 6. The dominant frequency varied in dependence on the sodium flow rate from 1.5 to 8.8 Hz. The application of the results to the ZEUS measuring system shows that the system may well be used for measuring leaks of water into sodium in the order of 10^{-1} g/s in a steam generator of the modular type. The results obtained allow the assumption that the electromagnetic flowmeter will be able to operate as a sensor for detecting water leaks into sodium with a sensitivity as high as 10^{-2} g/s.

- 3724 VIBRATIONAL ANALYSIS OF THE SUPERPHENIX INTERNAL SHELLS. Livolant, M.; Jeanpierre, F. (CEA Centre d'Etudes Nucleaires de Saclay, 91 = Gif-sur-Yvette (France)). Nucl. Eng. Des.; 47: No. 2, 289-296(May 1978). (In French).

The geometry of the internal shells of Superphenix is somewhat complicated: they are mainly axisymmetric with some special parts, like pumps and heat exchanger crossing, non-axisymmetric. The flow leaving the reactor core or the heat exchanger strikes some of these shells and induces vibrations. Owing to the expected long life of the plant, without the possibility of intervention on the shells, it is necessary to verify that the vibrational level is sufficiently low and that there is no risk of fatigue. A first estimate of the vibrational level has been calculated. This paper explains the formalism used and describes the main steps of the calculation: (a) estimation of the fluctuating pressures; (b) calculation of the structural resonances, taking into account the sodium effects; and (c) estimation of the resulting vibrational amplitude. A short review of the experimental work planned to verify and validate the calculation results is presented.

- 3725 SIMULATION OF THE DYNAMIC CHARACTERISTICS OF STEAM GENERATOR FOR FBR PLANT. Kawamata, K. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center); Morimoto, H.; Maruyama, H.; Hida, K.; Hakogami, T. Mitsubishi Juko Giho; 15: No. 3, 318-325(May 1978). (In Japanese).

The development of the liquid metal fast breeder reactor (FBR) in Japan has been pushed forward as a national project under the guidance of the Power Reactor and Nuclear Fuel Development Corporation (PNC) who is charged with the responsibility for this program. The

project is aimed at constructing the "Monju", 300 MWe prototype FBR. As the steam generator is one of the most important components of the thermal transportation system of the FBR plant, we have put emphasis on the research and development of a more efficient steam generator. As the manufacture of the components of the FBR plant we have been making the analysis and evaluation of the dynamic characteristics of the steam generator by mathematical simulation. In order to improve the performance, safety, reliability and utility rate of the whole plant by the sure operation of the steam generator, these analysis and evaluation should be as accurate as possible, for which it is necessary to know the dynamic characteristics of the steam generator and verify the accuracy of simulation program. This paper reports on the results of our verification of the accuracy of the simulation program developed by our company by the comparison of the calculated results with the results of tests on the dynamic characteristics of the 50 MW No. 2 steam generator for "Monju" R and D which our company designed, manufactured and delivered to the PNC Oarai Engineering Center. As the result of this comparison it has been confirmed that the values obtained by use of this program are in good agreement with experimental values concerning the dynamic characteristics of the steam generator under high, low and very low load conditions and that simulation can be made in a practically short calculation time (CPU time).

3726 DECAY HEAT REMOVAL TEST OF SPENT FUEL IN "MONJU" EX-VESSEL TRANSFER MACHINE, PHASE 2. Inoue, T. (Fuji Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). FAPIG (Tokyo); No. 89, 2-6, 12(May 1978). (In Japanese).

Concerning decay heat removal of spent fuel assemblies in the ex-vessel transfer machine for the prototype fast breeder reactor "Monju", the construction of which is being planned in the Power Reactor and Nuclear Fuel Development Corporation, indirect cooling test has been made on the mock-up fuel assembly in a pot containing sodium. From the test, many useful data for the design of the actual ex-vessel transfer machine was able to be acquired. In this way, the validity of a thermal calculation code for the design was also examined, and its practicability was confirmed. The indirect cooling test and its results and the analysis including that on code validity are described. (1) The emissivity on the outside surface of the pot is 0.5 - 0.8. (2) The maximum temperature on fuel-pin surfaces in sodium flow with natural convection by the calculation code was in good agreement with the test results. (3) Not only the natural convection inside and outside the fuel assembly but also local natural convection above the heat generating part contribute to the heat removal.

3727 SUPER-PHENIX: FIRST STEP IN THE COMMERCIAL DEVELOPMENT OF BREEDER REACTORS. Banal, M. (Electricite de France, 75 - Paris); Megy, J. (CEA, 75 - Paris; France)). Ann. Mines; 184: No. 5-6, 127-140(May-Jun 1978). (In French).

Super-Phenix is an indispensable step in a program for the development of the breeder reactors. Its power and technical options are recalled. The administrative directives - particularly concerning safety - are in agreement with the regulation enforced in France relative to nuclear power stations and work has begun on the Creys-Malville site within the international framework where the

development of breeders in Europe now finds its place. The saving in uranium due to breeders is certain, but their cost prices remain to be proved. The construction of Super-Phenix is the proof that it is possible to rely on the breeders in the final energetic results for the future, but it is of a type which can still be modified to a considerable extent, especially in order to reduce investments.

3728 FUEL CONFIGURATION FOR A NUCLEAR REACTOR COOLED WITH LIQUID METAL. Duncombe, E.; Gratton, C.P.; Adamson, J. (to UKAEA London Office; Deutsches Patentamt, Muenchen (Germany, F.R.)). German(FRG) Patent 2,746,555/A/. 20 Apr 1978. 15p. (In German).

In a liquid metal-cooled reactor there are provided openings along the coolant channels in order to reduce local thermal peaks. At the outlet of the coolant channel a transducer is monitoring the cooling of the fuel. The heat is removed by several open heat transfer pipes distributed over the cross-section of the coolant channel.

3729 IMPROVEMENTS IN OR RELATING TO PIPE JOINTS. Lee, A.S. (to English Electric Co. Ltd., London (UK)). British Patent 1,507,948/A/. 19 Apr 1978. 9p.

Pipe joints are described that are particularly suitable for liquid metal cooled nuclear reactors. The object is to provide a joint capable of accommodating movements resulting from differential expansion of the reactor components. Full constructional details are given.

3730 TUBE IN SHELL HEAT EXCHANGERS. Linning, D.L. (to UKAEA). British Patent 1,507,833/A/. 19 Apr 1978. 3p.

An improved tube-in-shell heat exchanger is described for use as a boiler for generating steam by heat exchange with liquid metal. It comprises a bundle of spaced tubes that conduct the liquid metal between headers and that extends through a shell containing water, the ends of this shell being closed by tube sheets spaced from the headers. The heat exchange tubes have internal thermally insulating sleeves each of which comprises a pair of spaced co-axial tubes sealed together at each end to form a closed interspace that contains inert gas. In this arrangement the heat flux between the tubes and tube sheet adjacent the inlet header for liquid metal is greatly reduced and the deposition of salts on the water side of the tube sheet and on the outside of the tubes in the vicinity of the tube sheet is thereby greatly reduced.

3731 ON-LINE ULTRASONIC GAS ENTRAINMENT MONITOR. Day, C.K.; Pedersen, H.N. (to Dept. of Energy). US Patent 4,083,225. 11 Apr 1978. Filed date 17 Mar 1976. 6p. PAT-APPL-667,924.

Apparatus employing ultrasonic energy for detecting and measuring the quantity of gas bubbles present in liquids being transported through pipes is described. An ultrasonic transducer is positioned along the longitudinal axis of a fluid duct, oriented to transmit acoustic energy radially of the duct around the circumference of the enclosure walls. The back-reflected energy is received centrally of the duct and interpreted as a measure of gas entrainment. One embodiment employs a conical reflector to direct the transmitted acoustic energy radially of the duct and redirect the reflected energy back to the transducer for reception. A modified embodiment employs a

cylindrical ultrasonic transducer for this purpose.

- 3732 **POLITICS OF REPROCESSING.** Barnaby, F. (Stockholm International Peace Research Inst. (Sweden)). New Sci.; 78: No. 1097, 18-19(6 Apr 1978).

The political and economic questions surrounding the use of plutonium as a nuclear fuel are examined. The advantages of liquid-metal fast breeder reactors are considered against the background of the sharp decline in demand for nuclear power. However the very real problems of their use with respect to the possible proliferation of nuclear weapons are discussed and the use of plutonium from fuel elements removed from the core and the breeder blanket in the fabrication of nuclear weapons is considered. The part played by reprocessing of spent fuel elements from ordinary reactors for use in breeder reactors and possible proliferation restrictions imposed in countries with no such reprocessing facilities are mentioned.

- 3733 **THERMAL POWER PLANT.** Mangus, J.D. (to Westinghouse Electric Corp., Pittsburgh, Pa. (USA)). German (FRG) Patent 2,741,945/A/. 6 Apr 1978. 24p. (In German).

The patent claims a method to improve intermediate superheating in nuclear power plants with liquid-metal-cooled reactors, so that temperature changes in the steam led from the high-pressure turbine to the intermediate superheater can be taken up.

- 3734 **FUEL SUBASSEMBLY LEAK TEST CHAMBER FOR A NUCLEAR REACTOR.** Divona, C.J. (to Dept. of Energy). US Patent 4,082,607. 4 Apr 1978. Filed date 30 Sep 1976. 4p. PAT-APPL-728,360.

A container with a valve at one end is inserted into a nuclear reactor coolant pool. Once in the pool, the valve is opened by a mechanical linkage. An individual fuel subassembly is lifted into the container by a gripper; the valve is then closed providing an isolated chamber for the subassembly. A vacuum is drawn on the chamber to encourage gaseous fission product leakage through any defects in the cladding of the fuel rods comprising the subassembly; this leakage may be detected by instrumentation, and the need for replacement of the assembly ascertained.

- 3735 **DETERMINATION OF REACTIVITY FROM POWER SPECTRAL DENSITY MEASUREMENTS WITH ^{252}Cf .** Mihalcz, J.T.; Pare, V.K.; Ragan, G.L.; Mathis, M.V.; Tillett, G.C. (Oak Ridge National Lab., TN). Nucl. Sci. Eng.; 66: No. 1, 29-59(Apr 1978).

The theory of a method of determination of reactivity from power spectral density measurements with ^{252}Cf and the results of experiments with a critical assembly mockup of a liquid-metal fast breeder reactor (LMFBR) and with uranium (93.2 wt % ^{235}U) metal cylinders and a sphere are presented. This method of reactivity determination has an advantage over existing methods in that it determines the reactivity only from properties of the reactor at the subcritical state of interest and thus does not require a calibration near delayed criticality. In these experiments, the reactivity was varied by changing the fissile loading or the amount of neutron absorber inserted; for the LMFBR mockup, the reactivity varied to approximately 75 dollars subcritical, and for the uranium metal assemblies to approximately 30 dollars subcritical. These

experiments verified for the first time the predictions of theory that could be tested in the measurements. This method has potential use in the fuel loading of reactors to determine the reactivity far subcritical before initial criticality is achieved. It has the advantage of not requiring a calibration at known reactivity by another method; furthermore, the interpretation of the measured data to obtain the reactivity does not depend on relative or absolute values of the source intensity or detection efficiency. It can also be used to determine the reactivity of assemblies where loading to criticality is undesirable or where sufficient material to achieve criticality is not available.

- 3736 **CALCULATION OF THE REACTIVITY DUE TO BUBBLE COLLAPSE WITH THE PROBABILITY TABLE METHOD.** Hoffman, T.J.; Petrie, L.M. (Oak Ridge National Lab., TN). Nucl. Sci. Eng.; 66: No. 1, 67-74(Apr 1978).

The probability table method, originally developed for unresolved resonance calculations, is adapted to the calculation of bubble worths in a reactor core. Unlike previous approaches to this problem, this method is applicable to a broad class of problems, avoids a high variance Monte Carlo calculation, and preserves the main features of a random bubble distribution. This approach is illustrated by the solution to several problems of varying degrees of complexity.

- 3737 **PROBABILISTIC ACCIDENT ANALYSIS WITH DETERMINISTIC CODES USING STOCHASTICALLY VARIABLE PARAMETERS.** Abramson, P.B.; Hummel, H.H.; Gelbard, E.M.; Pizzica, P.A.; Sienicki, J.J. (Argonne National Lab., IL). Nucl. Sci. Eng.; 66: No. 1, 14-23(Apr 1978).

In the use of large computers to analyze severe accidents in liquid-metal fast breeder reactors (LMFBRs), it has long been recognized that many of the fundamental phenomena cannot be precisely predicted because of uncertainty in the parameters that govern them. As a direct result, mechanistic analysis of such accidents has proceeded along a parametric path in which these variables are fixed at a certain constant value for the entire calculation. The influence of variation of this value is assessed by making a series of complete calculations with the parameter set at a different value for each such element of the series. While some parameters may be thought of as 'correlated' or fixed for an entire calculation, very few are in fact constant throughout a reactor, and many are (for practical purposes) nearly completely uncorrelated, either in space or time, during the hypothetical accident. Thus, such analysis has created a set of results that are not indicative or representative of an accident involving uncorrelated or only partially correlated variable parameters. A methodology is described for dealing with various degrees of uncertainty or incoherence in these parameters. By using two very different mechanistic codes (FX2-PQOL and EPIC), we demonstrate that the treatment of uncorrelated parameters, such as droplet/particle size in a hypothetical core disruptive accident, as random variables with a certain probability distribution during each complete calculation of a series of calculations produces as much as an order of magnitude less uncertainty in the end result than had been obtained assuming perfect correlation. A small list of parameters is categorized as either correlated or uncorrelated for some of the other LMFBR accident analysis codes. The technique can be easily implemented in a broad spectrum of accident analysis codes with

similar benefits.

- 3738 EXPERIMENTAL INVESTIGATION OF GAS EVOLUTION AS THE MECHANISM FOR FREE-CONTACT FRAGMENTATION. Shiralkar, G. (Univ. of California, Berkeley); Rohsenow, W.; Sonin, A.; Todreas, N. Nucl. Sci. Eng.; 66: No. 1, 103-109(Apr 1978).

An experimental investigation was made to assess the hypothesis of Epstein that violent release of dissolved gas upon rapid cooling of a molten metal is responsible for observed free-contact fragmentation. Results from quenching tin drops initially heated in vacuum and air to develop drastic initial differences in oxygen solubility levels did not demonstrate any differences in observed fragmentation occurrence or intensity. It is concluded that this hypothesis is not the operative mechanism for free-contact fragmentation in the tin-water system.

- 3739 (IAEA-CN--36/342) SAFETY OF SODIUM-COOLED FAST REACTORS. Bagdasarov, Yu.E.; Matveev, V.I.; Kuznetsov, I.A.; Krasnoyarsov, N.V.; Kochetkov, L.A.; Kulakovskiy, M.Ya.; Pomerantsev, G.B. Apr 1978. Translation of USSR paper (CONF-770505--54(trans)). 21p. Dep. NTIS, PC A02/MF A01.

From International conference on nuclear power and its fuel cycles; Salzburg, Austria (2 May 1977).

Some physical and technical aspects of sodium-cooled fast reactor safety are discussed. An approach to the solution of safety problems used in the designs of the USSR first fast-reactor generation was based (as a result of thermal-reactor operating experience) on a logical analysis of the tree of failures, on careful development and testing of safety systems and devices, and on duplication and echelonment of the most important of them. A review of main works carried out for the last two decades is given and the most important results of design studies, research and operating experience obtained on fast reactors as to accumulation of data and characteristics in the area of safety are presented.

- 3740 (IAEA-CN--36/356) USSR FAST REACTOR DEVELOPMENT PROGRAM AND ITS CURRENT STATE. Kazachkovski, D.D.; Meshkov, A.G.; Mitenkov, F.M. Apr 1978. Translation of USSR paper, 1977 (CONF-770505--190(Trans.)). 24p. Dep. NTIS, PC A01/MF A01.

From International conference on nuclear power and its fuel cycles; Salzburg, Austria (2 May 1977).

Fast power reactors make it possible to use in a more efficient way existing natural resources of nuclear raw materials, ensuring by that large-scale development of nuclear power. The state of fast reactor development work in the USSR and some questions of fuel cycle are discussed.

- 3741 DIFFUSION AND TRAPPING OF TRITIUM IN GRAIN BOUNDARIES OF 304L STAINLESS STEEL. Abraham, P.M.; Elleman, T.S.; Verghese, K. (North Carolina State Univ., Raleigh (USA). Dept. of Nuclear Engineering). J. Nucl. Mater.; 73: No. 1, 77-88(Apr 1978).

Permeation rates of tritium through 304L stainless steel are calculated using a grainboundary diffusion model and measured values of bulk and grainboundary diffusion coefficients. Computed results show that the tritium release rate through grainboundaries should be significantly greater than through the lattice. These predictions, however, are

not borne out by permeation experiments. In order to explain this discrepancy, autoradiographic examinations were carried out. The results show that although grainboundaries are indeed fast diffusion paths, their contribution to the overall permeation rate may be minimized by trapping in surface oxide films and possibly by low solubility of tritium within the grainboundaries.

- 3742 EFFECT OF ION-PLATED MOLYBDENUM ON CORROSION OF V/MO ALLOYS IN LIQUID SODIUM. Iwao, N.; Suzuki, T.; Fukutomi, M.; Okada, M.; Watanabe, R. (National Research Inst. for Metals, Tokyo (Japan)). J. Nucl. Mater.; 73: No. 1, 105-110(Apr 1978).

The effect of ion-plated molybdenum of less than 5 μm in thickness on corrosion of V-Mo alloys has been examined in high purity liquid sodium at 700°C for 500 h. The oxygen concentration in the sodium was about 1.0 ppm ('Vanadium Wire Equilibration Method'). By molybdenum coating, the oxygen absorption from sodium was remarkably suppressed and the dissolution of solid metal into sodium was reduced to less than 10 percent of that for the uncoated alloy. The tensile properties at room temperature, the fractography and the microhardness have shown good promise for resistance of the molybdenum coating to corrosive attack by sodium.

- 3743 NEW NUCLEAR POWER PLANTS IN THE FEDERAL REPUBLIC OF GERMANY 1978. Atomwirtsch., Atomtech.; 23: No. 4, 164-177(Apr 1978). (In German).

A survey in tabular form of all nuclear power plants in operation up to March 1978 is followed by various chapters describing the main data of the 12 power reactors now under construction. Finally, 11 nuclear power plant projects are described.

- 3744 ASSESSMENT OF THE PERFORMANCE OF UK CFR CANDIDATE STRUCTURAL MATERIALS. Charnock, W.; Marshall, P. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Nucl. Energy; 17: No. 2, 119-132(Apr 1978).

The service performance and the extent of degradation of properties of the main UK commercial fast reactor candidate structural materials AISI 316 and 9 Cr-1 Mo over a 30 year lifetime in a sodium cooled reactor are assessed. From the point of view of creep and fatigue behaviour the following effects are judged to be unlikely to give rise to problems either at plain surfaces or at crack tips: substitutional element transfer; ferrite layer formation; liquid metal embrittlement; tertiary creep embrittlement; carbon transfer in primary and secondary circuits; oxygen effects at normal operating levels; and self welding. Much of the information required for design may be obtained in air or in inert gas environments. The self-welding phenomenon, however, has implications for seals, contacting surfaces, and demountable components. Important areas of uncertainty are as follows: thermal degradation of parent material welds, especially transition joints; caustic and very high oxygen concentrations arising from fault conditions; loss of toughness and ductility due to irradiation, particularly in welds; and localized carburization adjacent to pump leaks.

- 3745 OPTIMIZATION OF SODIUM COOLED FAST REACTORS. CONFERENCE REPORT. Grieve, M.S. (Central Electricity Generating Board, London (UK). Nuclear Health and Safety Dept.). Nucl. Energy; 17: No. 2, 80-84(Apr 1978).

A review is presented of the conference on Optimization of Sodium Cooled Fast Reactors, held by the British Nuclear Energy Society in London. Some of the more important topics and others of particular interest are reviewed. From the opening session, a view of the optimisation approach of different countries is obtained. Other papers are considered under the topics of safety; core, subassemblies and fuel; overall design; sodium circuits and steam plant; and system benefit. Heterogeneous cores were the subject of some divergence of opinion in the closing session.

3746 CLAYMORE CODE AND ITS USE FOR DESCRIBING COOLANT BOILING IN LMFBR SUBASSEMBLIES.

Gregory, C.V.; Lord, D.J. (UKAEA, Dounreay. Nuclear Power Development Establishment). Nucl. Energy; 17: No. 2, 107-114 (Apr 1978).

The interest in subassembly incidents as an important factor in liquid metal fast breeder reactor safety analysis has led to the development of the CLAYMORE code at the Dounreay Nuclear Power Development Establishment. The code contains a simple boiling model and, being derived from SABRE, is able to describe recirculating flows in subchannel geometries. The code is briefly described, particular attention being paid to the homogeneous boiling model and the method adopted for the inclusion of slip, and results from a range of calculations are presented. The results are used to demonstrate the influence on the description of boiling of three inherent uncertainties in the code, provided by lack of knowledge on inter-channel mixing, by the assumption of a bundle geometry unaffected by the temperature field, and, for blockage calculations, by the difficulty in assessing the magnitude of the leakage flow through the blockage.

3747 NUCLEAR POWER AND THE PROLIFERATION ISSUE. Marshall, W. (UKAEA). Atom (London); No. 258, 78-102 (Apr 1978).

This paper by the Deputy Chairman of the UK Atomic Energy Authority discusses what is termed nuclear proliferation on the basis of which problems are real and which are a misapprehension and suggests a way that will retain the benefits of nuclear power while providing a more certain safeguard against undesirable proliferation. Headings include Pu production and accessibility, the use of Pu, fast reactor fuel, and policy for the interim period until fast reactors are well established throughout the world. The author considers that the availability of Pu is not the most serious proliferation problem facing the world. Appendices include alternative routes for obtaining nuclear weapon materials, critical mass of Pu, the flow of Pu in the pressurised water and liquid metal fast breeder reactors.

3748 FEDERAL CONSTITUTIONAL COURT AFFIRMS ADMISSIBILITY OF DECISION IN THE MATTER OF THE FAST BREEDER KALKAR. Energiewirtsch. Tagesfragen; 28: No. 4, 265 (Apr 1978). (In German).

In the case of the examination of the constitutionality of section 7 Atomic Energy Act, in as far as this article enables the licensing of nuclear power plants of the type called Fast Breeder, the Second Senate of the Federal Constitutional Court has answered the OVG Muenster's motion to stay proceedings dated Aug 18th, 1977, with the following 'interim decision' - 2 B v L 8/77 - dated Jan 31st, 1978: 'The action is admissible'. The verdict was unanimous. The main grounds upon which the interim judgment is based are given in full.

3749 OPERATING EXPERIENCE AND SODIUM-WATER REACTION TESTING IN THE LARGE LEAK TEST RIG. DeBear, W.S.; Neely, H.H. (Liquid Metal Engineering Center, Canoga Park, CA). J. Eng. Power; 100: No. 2, 347-355 (Apr 1978).

DOE. LMFBR TYPE REACTORS; STEAM GENERATORS; FAILURE MODE ANALYSIS; MOLTEN METAL-WATER REACTIONS; TESTING; TEST FACILITIES; SODIUM

3750 IRRADIATION EFFECTS ON THE BEHAVIOR OF STRUCTURAL MATERIALS AT ELEVATED TEMPERATURE. Abo-El-Ata, M.M. (San Francisco State Univ., CA). J. Eng. Mater. Technol.; 100: No. 2, 121-127 (Apr 1978).

A qualitative review of the effects of fast neutron irradiation on the mechanical behavior of materials used in the construction of Fast Breeder Reactor Core Components is presented in three sections. First, effects of preirradiation on the strength, ductility, elastic moduli, thermal creep, creep-rupture, and fatigue are discussed. Second, the phenomenon of irradiation induced swelling is explained and the effects of temperature, cold work, and stress on swelling are presented. Third, irradiation creep is defined and the effects of temperature, stress, and cold work on irradiation creep are discussed. The review is based on the most recently available experimental data on austenitic stainless steels.

3751 FACING THE MATERIALS CHALLENGE IN FAST REACTOR STEAM GENERATOR TECHNOLOGY. Little, E.A.; Pugh, S.F. (UKAEA, Harwell. Atomic Energy Research Establishment). Nucl. Eng. Int.; 23: No. 270, 39-43 (Apr 1978).

Highlights are presented from an international conference on 'Ferritic steels for fast reactor steam generators' held in London, 1977. Papers covered all relevant materials aspects of chrome molybdenum steels, the main choice for fast reactor steam generators and super heaters being 2 Cr/Mo, 9Cr/Mo and 12Cr/Mo steels. The heat treatment response of the Cr/Mo steels, is such that it is possible to achieve a wide range of mechanical properties. Consideration is given to general metallurgy, operating experience, effect of composition on properties, corrosion in sodium, water-side corrosion, and fabrication processes.

3752 USE AND STATUS OF FAST BREEDER DEVELOPMENT. Brandstetter, A. (Internationale Natrium-Brutreaktor-Bau G.m.b.H., Bergisch Gladbach (Germany, F.R.)); Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). Waerme; 84: No. 2, 57-61 (Apr-Jun 1978). (In German).

The high boiling point of sodium accounts for its excellent safety characteristics permits conventional steam states, i.e. high plant efficiencies. It is wrong to describe sodium as an explosive medium. Sodium-water reactions and sodium fires will, at the most, reduce availability. They present a low danger potential for the operating personnel and no danger potential at all for the population in the neighbourhood of the plant. The operation of test facilities and power plant has shown that the state of the art of sodium technology permits reliable power plant operation.

3753 ACTUAL CONTROLLING OF STANDARDS CONCERNING THE 'FAST BREEDER REACTOR'. REMARKS ON THE DECISION OF THE OVG MUENSTER TO PRESENT A CASE TO THE FEDERAL CONSTITUTIONAL

COURT. Rengeling, H.W. (Hamburg Univ. (Germany, F.R.)). Off. Verwalt.; No. 8, 277-282(Apr 1978). (In German).

If the decision of the OVG Muenster to present the case to the Federal Constitutional Court asking whether Article 7 of the atomic energy law corresponds to the constitution or not, as far as the article allows the licensing of a fast breeder reactor, two problems arise: The legal question in how far the actual controlling of standards is to be preceded by a statement of facts given by the court of first instance, and the problem behind concerning the responsibility of decision. The Federal Constitutional Court should accept the responsibility of decision to be borne by it.

- 3754 ULTRA-SONIC FLOW METER FOR THE SODIUM COOLED FAST BREEDER REACTOR. Iida, S.; Araki, H.; Oda, M.; Matsunaga, Y.; Yoshiyama, Y. (Mitsubishi Electric Corp., Tokyo (Japan)). Keisoku Jido Seigyo Gakkai Ronbunshu; 14: No. 2, 178-183(Apr 1978). (In Japanese).

This paper describes an ultra-sonic flow meter (U.S. FM) designed to measure coolant flow rate of the fast breeder reactor power plant. The coolant is high temperature liquid sodium, the U.S. FM was improved to be useful in the high temperature environment. First, to eliminate the high temperature influence of the pipe to transducers, the guide rods were used. Second, the electronic circuits were improved to measure the flow rate by the difference in the transmission time of signals. Third, the electronic circuits were endowed with a capability of automatic temperature compensation and protected against bubble disturbance. With the improved U.S. FM, a study of acoustic transmission in liquid sodium was carried out. It was found that the sonic property of liquid sodium is similar to that of water. The U.S. FM was tested in a water flow to exhibit its operational characteristics including the linearity, the time constant and the bubble influence. The results showed up as expected and confirmed the promising aspects of the present measuring method.

- 3755 DEVELOPMENT AND PRODUCTION OF TUBES WITH INNER FINNING FOR NUCLEAR POWER. Starek, J.; Stanek, B. (Vyzkumny Ustav Hutnictvi Zeleza, Dobra (Czechoslovakia)). Strojirenstvi; 28: No. 4, 200-202(Apr 1978). (In Czech).

Increased heat transfer due to inner finning of tubes results in increasing the life of tubes in steam generators of sodium cooled reactors and in limiting unfavourable effects of temperature changes on the tubes. The development was therefore commenced of tubes with inner longitudinal and inner helical fins. The technology of drawing tubes on linear drawing benches using a short cylindrical mandrel was accepted as most suitable. The effect was studied of the geometry of tools and fixtures and the relationships were tested between the size of drawing reduction of the diameter and the initial wall thickness of different tubes. It was found that the increase in the fin height depended on the size of the wall thickness reduction. Tubes 18 mm in diameter, 3 mm in wall thickness and 0.5 and 1 mm in fin height, 2 mm in fin width and 5.5 to 7 m in length, and tubes with helical fins of 0.5 mm and 1 mm in fin height and with a pitch of 200, 250 and 300 mm were subject to tests confirming that for longitudinal fins heat transfer improved by 20% and for helical fins by 50%.

- 3756 IMPROVEMENT OF LIQUID-METAL FAST BREEDER REACTOR CHARACTERISTICS BY APPLYING A MAGNETIC

FIELD. Shmelev, A.N.; Ilyunin, V.G.; Murogov, V.M. At. Energ. (USSR); 44: No. 4, 359-360(Apr 1978). (In Russian).

FLOW RATE; FUEL ASSEMBLIES; HYDRAULICS; LMFBR TYPE REACTORS; MAGNETIC FIELDS; MECHANICAL VIBRATIONS; POWER DENSITY; SODIUM; TURBULENT FLOW

- 3757 NUCLEAR REACTOR FACILITY. Wampole, N.C. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)); Deutsches Patentamt, Muenchen (Germany, F.R.). German (FRG) Patent 2,742,009/A/. 30 Mar 1978. 13p. (In German).

In order to improve the performance of maintenance and inspections it is proposed for a nuclear reactor facility with a primary circuit containing liquid metal to provide a thermally insulated chamber, within which are placed a number of components of the primary circuit, as e.g. valves, recirculation pump, heat exchangers. The isolated placement permit controlled preheating on one hand, but prevents undesirable heating of adjacent load-bearing elements on the other. The chamber is provided with heating devices and, on the outside, with cooling devices; it is of advantage to fill it with an inert gas. (UWI) 891 HP.

- 3758 NUCLEAR REACTOR CORE COOLING ARRANGEMENT. Redding, A.H. (to Westinghouse Electric Corp.). US Patent 4,080,252. 21 Mar 1978. Filed date 18 Feb 1975. 8p.

A core cooling system for a nuclear reactor having a plurality of primary fluid flow systems is described. The reactor coolant flow from the primary systems is joined upon entering the pressure vessel. Jointure is accomplished in a common chamber causing high coolant flow velocities at low static pressures. If a pipe ruptures in one of the primary fluid flow systems, the low pressure in the common chamber minimizes leakage from the intact flow systems. This allows continuation of coolant flow through the nuclear core for a sufficient length of time to effectively eliminate the possibility of thermal damage.

- 3759 HORIZONTAL BAFFLE FOR NUCLEAR REACTORS. Rylatt, J.A. (to Dept. of Energy). US Patent 4,080,255. 21 Mar 1978. Filed date 31 Aug 1976. vp.

PAT-APPL-719,318.

A horizontal baffle disposed in the annulus defined between the core barrel and the thermal liner of a nuclear reactor thereby physically separating the outlet region of the core from the annular area below the horizontal baffle is described. The horizontal baffle prevents hot coolant that has passed through the reactor core from thermally damaging apparatus located in the annulus below the horizontal baffle by utilizing the thermally induced bowing of the horizontal baffle to enhance sealing while accommodating lateral motion of the baffle base plate.

- 3760 METHOD OF LOCATING A LEAKING FUEL ELEMENT IN A FAST BREEDER POWER REACTOR. Honekamp, J.R.; Fryer, R.M. (to Dept. of Energy). US Patent 4,080,250. 21 Mar 1978. Filed date 20 Nov 1973. 8p.

PAT-APPL-417,652.

Leaking fuel elements in a fast reactor are identified by measuring the ratio of ^{134}Xe to ^{133}Xe in the reactor cover gas following detection of a fuel element leak, this ratio being indicative of the power and burnup of the failed fuel element. This procedure can be used to identify leaking fuel elements in a power breeder reactor while continuing operation of

the reactor since the ratio measured is that of the gases stored in the plenum of the failed fuel element. Thus, use of a cleanup system for the cover gas makes it possible to identify sequentially a multiplicity of leaking fuel elements without shutting the reactor down.

- 3761 PLUG-TO-PLUG GAS TRANSFER SYSTEM. Poindexter, A.M. (to Dept. of Energy). US Patent 4,080,254. 21 Mar 1978. Filed date 28 Apr 1978. 8p.
PAT-APPL-680,826.

A system is described for conducting a fluid from one component to another component of a nuclear reactor wherein at least one such component is a rotatable closure head plug capable of movement relative to the other component. The conducting system utilizes the annulus located between the components as a connecting passageway for the fluid.

- 3762 CORE DISRUPTIVE ACCIDENT MARGIN SEAL. Garin, J. (to Dept. of Energy). US Patent 4,078,969. 14 Mar 1978. Filed date 13 Aug 1976. 10p.
PAT-APPL-714,221.

An apparatus is described for sealing the annulus defined between a substantially cylindrical rotatable first riser assembly and plug combination disposed in a substantially cylindrical second riser assembly and plug combination of a nuclear reactor system. The apparatus comprises a flexible metal member having a first side attached to one of the riser components and a second side extending toward the other riser component and an actuating mechanism attached to the flexible metal member while extending to an accessible location. When the actuating mechanism is not activated, the flexible metal member does not contact the other riser component thus allowing the free rotation of the riser assembly and plug combination. When desired, the actuating mechanism causes the second side of the flexible metal member to contact the other riser component thereby sealing the annulus between the components.

- 3763 SEALED HEAT ACCESS AREA ENCLOSURE. Golden, M.P.; Govi, A.R. (to Dept. of Energy). US Patent 4,078,968. 14 Mar 1978. Filed date 28 Jul 1976. 6p.
PAT-APPL-709,351.

A liquid-metal-cooled fast breeder power reactor is provided with a sealed head access area enclosure disposed above the reactor vessel head consisting of a plurality of prefabricated structural panels including a center panel removably sealed into position with inflatable seals, and other panels sealed into position with semipermanent sealant joints. The sealant joints are located in the joint between the edge of the panels and the reactor containment structure and include from bottom to top an inverted U-shaped strip, a lower layer of a room temperature vulcanizing material, a separator strip defining a test space therewithin, and an upper layer of a room temperature vulcanizing material. The test space is tapped by a normally plugged passage extending to the top of the enclosure for testing the seal or introducing a buffer gas thereinto.

- 3764 NUCLEAR REACTOR WITH SELF-ORIFICING RADIAL BLANKET. Bishop, A.A.; Weiss, E.H.G.; Engel, F.C. (to Westinghouse Electric Corp.). US Patent 4,077,835. 7 Mar 1978. Filed date 29 Apr 1975. 6p.

The peripheral blanket of a breeder reactor

requires a coolant flow rate which is a varying fraction of that of the central core region. A self-orificing blanket cooling structure which is characterized by a predominance of radial coolant flow, generated by the pressure difference across the blanket, is utilized to supply the necessary cooling. The blanket fuel assemblies are surrounded by perforated cans to allow for radial crossflow through the blanket region.

- 3765 MAINTENANCE DEVICE FOR LIQUID SEALS IN NUCLEAR REACTORS. Poindexter, A.; Ricks, H. (to Westinghouse Electric Corp., East Pittsburgh, PA (USA); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,738,634/A/. 2 Mar 1978. 33p. (In German).

As a maintenance device for liquid seals arranged in the annular gaps between rotating closure plates of lids of fast breeder reactors an ultrasonic transducer is proposed. It is to be mounted onto a probe in the tank lid. The oscillation emitted by the oscillating area of the ultrasonic transducer are to make undesired deposits detach from the sealing fluid and thus retain the wetting capability of the metal surfaces, thereby preventing the passage of radioactive gases through the seal. The use of this system allows even those parts of a nuclear reactor plant to be maintained to which access is difficult. The design of the probe and that of the system is described in detail.

- 3766 THERMAL EXPANSION WORTHS FOR A LIQUID-METAL FAST BREEDER REACTOR INFERRED FROM SMALL-SAMPLE REACTIVITY MEASUREMENTS. Wei-Teh Lee, B. (General Electric Co., Sunnyvale, CA); Kaiser, R.E.; Hitchcock, J.T.; Russell, C.S. Nucl. Sci. Eng.; 65: No. 3, 429-440 (Mar 1978).

An indirect experimental technique for determining the expansion coefficient was developed to provide uncertainty estimates for expansion coefficient calculations. This technique uses an R_z reactivity worth map synthesized from small-sample reactivity traverse measurements for major materials over the reactor core and blanket regions. The experimentally based expansion coefficients, representing the reactivity change due to uniform axial and radial expansion, are deduced by appropriately integrating measured worth profiles. This technique was evaluated in Phase A of the Zero Power Plutonium Reactor Assembly 5. Direct calculations of the expansion coefficients were performed, and results were compared with the experimentally determined values. The validity of the technique used to derive expansion coefficients from worth measurements was evaluated. It is concluded that the total expansion coefficients are reasonably well calculated; however, the calculated radial expansion coefficient was overestimated. Sources of possible systematic errors in the experimentally based values were studied. Based on the present experiment, an uncertainty of $\pm 20\%$ (90% level of confidence) on expansion calculations using ENDF/B-III data is estimated for a clean core configuration.

- 3767 PRODUCTION OF ^{232}U IN A 1200-MW(E) LIQUID-METAL FAST BREEDER REACTOR. Mann, F.M.; Schenter, R.E. (Hanford Engineering Development Lab., Richland, WA). Nucl. Sci. Eng.; 65: No. 3, 544-547 (Mar 1978).

Uranium-232 production has been calculated in a 1200-MW(e) liquid-metal fast breeder reactor as a function of position, time, and core composition. Whenever thorium is present in the core, large quantities of ^{232}U relative to uranium (approximately 1000 ppm) are

generated. For ^{232}U generated in the blanket, production is on the order of 50 ppm.

design of future commercial liquid-metal fast breeder reactors.

- 3768 RADIOLOGICAL ASSESSMENT OF REPROCESSING ADVANCED LIQUID-METAL FAST BREEDER REACTOR FUELS. Till, J.E.; Bomar, E.S.; Morse, L.E.; Tennery, V.J. (Oak Ridge National Lab., TN). Nucl. Technol.; 37: No. 3, 328-339 (Mar 1978).

Results of the assessment were compared with those for reference oxide fuel. Candidate advanced fuels analyzed included $[(\text{U,Pu})\text{C}]$ and $[(\text{U,Pu})\text{N}]$ with selected concentrations of ^{15}N . Core neutronics and designs appropriate to advanced-fueled LMFBRs were used with the ORIGEN computer code to calculate compositions of spent core and blanket fuel equivalent to 50 GW-y of electrical energy generation. Confinement factors, specific to each radionuclide released to the atmosphere at the reprocessing plant, were used to calculate source terms describing the emission of radionuclides. Radiological impact within a 80-km radius of the facility was determined with the AIRDOS-II computer code, while the dose to the world population from ^{14}C released at the reprocessing plant was estimated with a multicompartiment global carbon cycling model. For carbide fuel, the calculated dose commitment to the total body of the maximally exposed individual was approximately 2.8 mrem. Corresponding dose commitments when nitride fuels were substituted ranged between 59 and 3.4 mrem, as the ^{14}N content in fuel was varied from 99.64%, the natural abundance of ^{14}N , to zero (i.e., 100% enrichment with ^{15}N), respectively. Dose commitment to the world population from ^{14}C produced in nitride fuel made with natural nitrogen may prohibit use of this fuel from an environmental standpoint. However, a combination of ^{15}N enrichment in the fuel and confinement of at least part of the ^{14}C at the reprocessing plant can significantly reduce both the amount of ^{14}C ultimately released to the atmosphere and the global impact from ^{14}C . It was concluded that no major differences exist in the radiological impact when carbide fuel is substituted for oxide fuel at the reprocessing plant. The use of nitride fuel may, however, require substantial enrichment with ^{15}N or significant improvement in effluent treatment techniques proposed for ^{14}C retention. 7 tables, 5 figs.

- 3769 CORE EXIT INSTRUMENTATION OF THE CLINCH RIVER BREEDER REACTOR PLANT. Carelli, M.D. (Westinghouse Electric Corp., Madison, PA). Nucl. Technol.; 37: No. 3, 261-273 (Mar 1978).

Assembly exit thermocouples are chosen for the Clinch River Breeder Reactor Plant as the instrumentation providing the most useful information at the minimum cost. One thermocouple is positioned at the exit of each fuel assembly and at approximately half of the radial blanket assemblies. The number of thermocouples, their positions, and characteristics are selected to satisfy the reactor control, surveillance, and design verification functions. The various uncertainties affecting the assemblies' coolant exit temperature measurements are quantitatively defined to correlate the measured temperature with the fuel rod design cladding temperature, which is the major parameter in determining the allowable fuel rod burnup and lifetime. Thus, appropriate factoring of thermocouple measurements allows the fuel assembly burnup to be increased quite significantly, with related cost savings of hundreds of millions of dollars. Due to the tremendous economic leverage on operating costs over the plant lifetime, close attention to proper instrumentation should be paid in the

- 3770 ELIMINATION OF TRANSURANIUM ELEMENTS BY BURNUP IN A POWER FAST BREEDER REACTOR. Oliva, G. (Univ. of Rome); Palmiotti, G.; Salvatores, M.; Tondinelli, L. Nucl. Technol.; 37: No. 3, 340-352 (Mar 1978).

The elimination of transuranium (TRU) elements by neutron absorption has been investigated in an actual power liquid-metal fast breeder reactor (LMFBR) (of the Superphenix type). Special fuel elements containing TRU oxides were considered in different core locations. The effects on design parameters have also been evaluated. The results show that the advantages of TRU elimination by means of LMFBRs, compared with using a thermal reactor, consist mainly of the small perturbation of the integral properties and design parameters of the reactor for the large amount of TRU that can be introduced. However, from the point of view of the TRU transmutation reaction rates, thermal reactors seem to be better. The choice of a compromise between the variation of design parameters and the TRU amount to be transmuted depends on the actual reactor design.

- 3771 COVA PROGRAMME FOR THE VALIDATION OF COMPUTER CODES FOR FAST REACTOR CONTAINMENT STUDIES. Hoskin, N.E. (UKAEA Weapons Group, Aldermaston. Atomic Weapons Research Establishment); Lancefield, M.J. (UKAEA Reactor Group, Winfrith. Atomic Energy Establishment). Nucl. Eng. Des.; 46: No. 1, 17-46 (Mar 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, USA (15 - 19 Aug 1977).

The UKAEA and the Joint Research Centre, Euratom, Ispra are engaged in a collaborative experimental programme carrying out a series of small scale, well instrumented tests aimed at providing high quality data of the stresses, strains and loads occurring when a well characterized source is released within a fluid in a containment vessel. In the UK the data are being used to validate the computer codes ASTARTE and SEURBNUK which are used in studies of the response of the fast reactor primary containment system in the event of a hypothetical reactor excursion. This paper describes the UK experimental programme and the development of the low density explosive which is used as the energy source; the rationale of the code development programme is presented together with a report on the progress that has been made in validating the two codes against the experimental data.

- 3772 STATUS REPORT ON ULTRASONIC TESTING OF AUSTENITIC WELDS. Herberg, G.; Laufer, W. Materialpruefung; 20: No. 3, 120-124 (Mar 1978). (In German).

Experience in the field of austenite systems has shown that surface and irradiation testing are not sufficient on their own to judge welded joints. Particularly in consequence of the development of Na-cooled breeder reactors, it appears necessary to demand the same expenditure for fabrication and iterative tests as for ferrite components for light water reactors (LWR). As the primary and secondary circuit components of Na breeders are fabricated from austenite materials, the ultrasonic (US) testing must also be adapted to the coarse-grained directional solidifying microstructure of austenite welded seams. World-wide development works are being carried out on this subject. The procedure here is not uniform.

- 3773 LMFBR--AN INTERNATIONAL EFFORT. Horst, K.M. (GE, Sunnyvale, Calif.). IEEE Trans. Power Appar. Syst.; PAS-97: No. 2, 529-535 (Mar-Apr 1978).

Major programs are underway in the industrialized nations to develop the Liquid Metal Fast Breeder Reactor (LMFBR) for application in their electric utility supply systems. In addition to the United States, these nations include France, the United Kingdom, the Federal Republic of Germany, Japan and the USSR. Belgium and The Netherlands have joined forces with Germany, and Italy is participating with France. Each of these nations has major laboratory facilities dedicated to the development of supporting technologies. They also have been building a series of reactors to provide the essential steps leading to widescale application. The purpose of this paper is to review the international progress which has already been made in the development of the LMFBR and to identify selected technical features of similarity and contrast. Note will also be made on the operating experiences to date and projections for the future. 19 refs.

- 3774 SODIUM BOILING EXPERIMENTS IN THE FRAME OF FAST REACTOR SAFETY STUDIES. Cenni, R.; Meneghello, S.; Ricci, F.; Romagnoli, M.; Savoni, O.; Scavano, G.; Sirtori, L. (CNEN, Cent Stud Nucl, Casaccia, Rome, Italy). Energ. Nucl. (Milan); 25: No. 3, 123-134 (Mar 1978). (In Italian).

This article presents the sodium boiling experiments recently performed at the Centro Studi Nucleari della Casaccia (Rome) in the frame of the fast reactor safety studies being carried out by CNEN. In particular, it describes three types of experiments, performed on test-sections mounted on the ENA-2 loop, in order to study the influence of the heat flux axial distribution, single phase transient and radial temperature profile on boiling dynamics. A preliminary analysis allows a rough extrapolation of the results obtained on a seven-rod test section to the case of a fast reactor fuel element. 10 refs.

- 3775 (JAPFNR--306) SODIUM OXIDE AEROSOL BEHAVIOR IN A CLOSED CHAMBER UNDER THERMAL CONVECTION FLOW. Nishio, G.; Kitani, S.; Takada, J. (Japan Atomic Energy Research Inst., Tokai, Ibaraki. Tokai Research Establishment). Mar 1978. Translation of PNC-J--250-75-14 Tr 29p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

The present work was undertaken to obtain the effect of thermophoretic adherence of nuclear aerosol on the reactor containments in the event of a hypothetical accident of LMFBR. In the accident, sodium oxide aerosol containing PuO_2 -- UO_2 fuel and fission products may be suspended in the containment and tends to deposit on the cold wall of the containment, so that the aerosol concentration decreases by thermophoresis in addition to settling and diffusional plating. Experiment was made to determine the relation for the degree of thermophoretic adherence of sodium oxide aerosol in a chamber. The results obtained by the experiment were compared with those of the calculation.

- 3776 DESIGN OF THE CORE CATCHER SYSTEM FOR THE FAST BREEDER REACTOR SNR 300. Keintzel, G. (Steinmueller (L. u. C.) G.m.b.H.,

Gummersbach (Germany, F.R.)); Rothfuss, H. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). Kerntechnik; 20: No. 3, 127-134 (Mar 1978). (In German).

The core catcher system of the sodium-cooled fast breeder reactor SNR 300 is intended to limit the consequences of a hypothetical core-melt accident and, as such, is relevant from the safety viewpoint. This paper describes the functions, concept and design of this system and presents a few examples illustrating the extreme conditions for which this system is designed.

- 3777 SELECTION OF THE MAIN CIRCULATION PUMPS FOR THE FIRST LOOP OF PROMISING PLANTS WITH SODIUM HEAT CARRIER. Kostin, V.I.; Kuropatov, A.I. Teploenergetika (Moscow); No. 3, 54-57 (Mar 1978). (In Russian).

Problems of selection of the flow-through part of the pump are considered. An evaluation of the mass and dimensional characteristics of several compared pumps is given. Economic efficiency of several variants of the design of a first-loop pump envisaged for promising commercial nuclear power plants with fast reactors is analyzed. Based on this analysis, conclusions are drawn on the selection of the optimal pump variant from the point of view of its characteristics and the minimal costs of a set of pumps for the plant.

- 3778 STUDIES ON TEMPERATURE-CONTROLLED POOL-BOILING HEAT TRANSFER. Sakurai, A.; Shiotsu, M. (Kyoto Univ, Inst of At Energy). Tech. Rep. Inst. At. Energy, Kyoto Univ.; No. 175, vp (Mar 1978).

A new computer-controlled experimental system to realize stable temperature-controlled operation in nucleate, transition and film boiling regions was developed. Characteristics of temperature-controlled pool-boiling heat transfer covering nucleate, transition and film boiling regions were studied for a platinum wire heater in water at atmospheric pressure. Effects of subcooling, heater orientation, heater dimension, heater surface contamination and of surface active agent were also studied. Thermal hysteresis effects due to increasing and decreasing heater temperature were found, for the first time, both in the nucleate boiling region and in the transition boiling region near the maximum heat flux point. Maximum heat flux of nucleate boiling and maximum heat flux of transition boiling were not the same. Artificial contamination on the heater surface and slight addition of surface active agent in water both severely influenced the transition boiling heat transfer. It was confirmed that the transition boiling was the combination of nucleate and film boiling. These results have application to once-through boilers forming a part of sodium cooled power reactors. 15 refs.

- 3779 PARLIAMENTARY POWERS CONCERNING GUIDING POLITICAL DECISIONS. DECISION OF THE OVG MUENSTER TO PRESENT THE LICENSING CASE OF THE KALKAR FAST BREEDER TO A HIGHER COURT. Kisker, G. Z. Parlam.; 9: No. 1, 53-62 (Mar 1978). (In German).

The author concludes that the step towards a plutonium industry would result in damage of unknown proportions if the forecast risks accumulate to an accident. For the author, the engagement of the Federal Diet seems to be therefore indispensable. The decision on the acceptance and restraining of such a risk should not be made in the offices of regional authorities without discussing the pros and

cons in parliament, i.e. in public.

- 3780 SODIUM VAPOR DEPOSITION ONTO A HORIZONTAL FLAT PLATE ABOVE LIQUID SODIUM SURFACE, (3). CONCENTRATION AND DIAMETER DISTRIBUTIONS OF SODIUM MIST IN COVER GAS. Kudo, K. (Hitachi Ltd., Ibaraki (Japan). Hitachi Research Lab.); Hirata, M. Nippon Kikai Gakkai Rombunshu; 44: No. 379, 1025-1035 (Mar 1978). (In Japanese).

Sodium vapour and sodium mist in the cover gas of a sodium system of a fast breeder reactor caused various problems. In this report, with the results of measurements of sodium mist concentration, the distribution of sodium mist diameter in cover gas was analytically obtained. The analysis was made by using the different nucleus model B. The measurement of the concentration of sodium mist was carried out with a sodium mist pot designed by the author. The experiment was done at the sodium temperature of 400 and 500 degree centigrade. The relations among sodium temperature, upper wall temperature, and the sodium mist concentration in cover gas were obtained. Evaluation of effective condensed nuclear radius in the cover gas was made by the comparison of analysis and experimental results. The results of this evaluation shows the following conclusions. It is impossible to express the distribution of sodium mist diameter by normal distribution or logarithmic normal distribution. Drop of sodium temperature results in the decrease of weight mean radius of generated sodium mist. Drop of upper wall temperature causes the decrease of weight mean radius, and increases sodium mist concentration.

- 3781 MOLTEN FUEL-SODIUM INTERACTION IN FAST REACTOR ACCIDENTS. Suchomel, J.; Adamik, V. (Vyskumny Ustav Energeticky, Jaslovske Bohunice (Czechoslovakia)). Jad. Energ.; 24: No. 3, 81-86 (Mar 1978). (In Czech).

A survey is presented of the problems of molten fuel-sodium interaction in model accidents of fast reactors with the occurrence of rapid heat transfer and the resulting strong pressure pulses. The INTER computation model is described which emphasizes the heat transfer and the CEFRA model which is focused on the significant influence of the dynamics of structures on the course of the interaction. The results of molten fuel-coolant interaction computations are given carried out for the fuel assembly and for the whole fast reactor core.

- 3782 EVALUATION OF HEAT TRANSFER TUBE FAILURE PROPAGATION DUE TO SODIUM-WATER REACTION IN STEAM GENERATOR. Nei, H. (Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). J. Nucl. Sci. Technol. (Tokyo); 15: No. 3, 192-199 (Mar 1978).

An evaluation was made of heat transfer tube failure propagation due to sodium-water reaction wastage in a sodium heated steam generator, by comparing an empirically derived wastage equation with leak detector responses. The experimental data agreed well with the wastage equation even for different values of distance-to-nozzle diameter ratio, though the formula had been based on wastage data obtained for only one given distance. The time taken for failure propagation was estimated for a prototype steam generator, and compared with the responses characteristics of acoustic detectors and level gages. It was found that there exists a range of leak rate between 0.5 and 100 g/sec, where the level gage can play a useful role in leak detection. The acoustic detector can be expected to respond more

rapidly than the cover gas pressure gage, if noise is kept below ten times the value observed in an experimental facility, SWAT-2.

- 3783 BOILING HEAT TRANSFER UNDER FORCED CONVECTION OF SODIUM. Sakurai, Akira; Serizawa, Akimi; Kataoka, Isao; Shiozu, Masahiro (Kyoto Univ., Uji (Japan). Inst. of Atomic Energy). Kyoto Daigaku Genshi Enerugi Kenkyusho Iho; 2: 24-26 (Mar 1978). (In Japanese).

As the basic research on the heat transfer concerning the safety of sodium-cooled fast neutron reactors, the authors have carried out the research on the boiling heat transfer under the natural convection of sodium. Moreover, the basic research on the non-boiling and boiling heat transfer under forced convection in steady and unsteady states is planned. As the unsteady state, the transient phenomena accompanying the rapid increase of heat generation of a heater and the rapid decrease of sodium flow rate are considered. First, as the research on the boiling heat transfer under the forced convection of sodium, the non-boiling heat transfer in the heat flux below the point of initial bubble generation was examined. It is the interesting problem to examine experimentally the heat transfer characteristics of sodium over the whole range of laminar flow, transition and turbulent flow regions. The experimental apparatuses and the experimental method are explained. The result of experiment is shown in relation to Nusselt number and Peclet number. In the experiment by the authors, the effect of heat conduction in axial direction was negligibly small. In liquid sodium also, in the transition region from laminar flow to turbulent flow, the tendency similar to the forced convection heat transfer obtained with mercury by Petukhov was shown. (Kako, I.)

- 3784 MATERIAL STUDY OF THE BOR-60 SHIM ROD THIMBLE IRRADIATED WITH 1.6×10^{23} N/CM² ($E > 0.1$ MEV). Krasnoselov, V.A.; Kosenkov, V.M.; Loboda, E.M.; Ostrovskii, Z.E.; Goncharenko, Yu.D. At. Energ. (USSR); 44: No. 3, 228-231 (Mar 1978). (In Russian).

A BOR-60 reactor shim rod thimble, made of the Kh18N9 austenitic steel with diffusion-chromized surface, has been investigated from the point of view of material science. It is established, that maximum radiation swelling of the material reaches approximately 15% at the distance of approximately 80 mm over the central plane of the reactor core at approximately 425 deg C. The thimble was irradiated with 1.5×10^{23} neut/cm² ($E > 0.1$ MeV) fluence. Thereby, discovered are significant differences in changing the volume and linear sizes of thimble faces. It is also established, that over the whole core length the thimble material has the zero plasticity level at 20 and 500 deg C test temperatures. Material embrittlement is suggested to relate not only with the irradiation effect, but with the structural changes in its interaction zone with sodium and, in particular, with Cr₂N phase formation, having a great parameter of volume discrepancy with austenite.

- 3785 PRIMARY AND SECONDARY SODIUM PURIFICATION SYSTEMS OF FAST REACTORS. Baklushin, R.P.; Kozlov, F.A. At. Energ. (USSR); 44: No. 3, 224-228 (Mar 1978). (In Russian).

Rational designing of sodium cleanup systems based on cold traps is investigated. Contributions of different sources to the total impurity burden of primary P₁ and secondary P₂

cooling circuits are evaluated using the operating experience of the BN-350 reactor. For 20 year lifetime period the impurity burden of P_1 will make up about 1400 kg of O_2 (approximately 5.5 t of impurities), the main share being due to core corrosion products. In the same lifetime the impurity burden of P_2 will amount to 3-14, mainly hydrogen compounds. Given are the method and results of calculations of an optimal number of a needed to cleanup the sodium. Currently available cold traps are noted to be sufficiently adequate in terms of productivity and capacity to ensure the purification of sodium cooled fast reactors up to 4000-5000 MW(th.).

- 3786 METHOD OF CONTINUOUS TESTING FOR LEAK-TIGHTNESS AND MECHANICAL RESISTANCE OF A REACTOR VESSEL JACKET AND A REACTOR IN WHICH SAID METHOD IS EMPLOYED. Cachera, P.C. (to Electricite de France (Service National)). US Patent 4,076,585. 28 Feb 1978. Priority date 7 Sep 1973, France, 4p.

The space formed between the primary vessel of a sodium-cooled fast reactor and a leak jacket surrounding the vessel is filled with an inert gas which is maintained at a pressure in the vicinity of atmospheric. A vacuum is created within the space formed between the leak jacket and the concrete reactor vault and means are provided for detecting the presence of inert gas within the evacuated space and thus continuously testing the jacket for leak-tightness and mechanical resistance. 4 claims, 1 figure.

- 3787 CONTROL FOR NUCLEAR REACTOR. Ash, E.B.; Bernath, L.; Facha, J.V. (to Rockwell International Corp.). US Patent 4,076,583. 28 Feb 1978. Filed date 4 Oct 1976. 10p.

A nuclear reactor is described having an inherently safe automatic shutdown capability comprising a pressure vessel, a core located within the vessel and containing a sufficient amount of fissionable material to establish and sustain a chain-type fission reaction, and a plurality of elongated conduits disposed in a regular arrangement within the vessel. Each of the conduits defines a fluid flow path and comprises a first section located within and extending through the core, and a second section located exteriorly of the core and positioned above the first section. Each of the conduits contains a plurality of substantially spherical bodies in a sufficient quantity to substantially fill the first section of each of the conduits. The conduits are provided with a first fluid connection to a lower part of the first section and a second fluid connection to an upper part of the second section for the passage of a pressurized fluid therethrough while maintaining the reactor in operation. The nuclear reactor further comprises a means for supplying the pressurized fluid to said first fluid connection for moving all of said plurality of bodies from said first section to said second section of each of the elongated conduits to form a hydraulically stacked bed of the spherical bodies. A bypass means is located intermediate the second fluid connection, and the second section of each of the conduits and includes a plurality of fluid passageways distributed along the length of the second section for permitting a portion of the pressurized fluid to bypass a portion of the stacked bed of bodies during normal operation of the reactor. 5 claims, 7 figures.

- 3788 FUSE AND APPLICATION OF SAID FUSE TO THE CONSTRUCTION OF AN EMERGENCY SHUTDOWN SYSTEM FOR A NUCLEAR REACTOR. Taulier, H.H.L.;

Brugelle, G. (to Electricite de France (Service National)). US Patent 4,076,587. 28 Feb 1978. Filed date 1 Mar 1976. 8p.

A fuse device for an automatic emergency shutdown system in fast reactors provides a coupling between a casing tube placed within a fuel can and a series of neutron-absorbing masses held together above the reactor core under normal operating conditions but released in free fall to the lower portion of the casing tube at the level of the reactor core as a result of melting of the fuse when operating characteristics such as temperature or neutron flux attain a level which exceeds a predetermined threshold.

- 3789 RODDED SHUTDOWN SYSTEM FOR A NUCLEAR REACTOR. Golden, M.P.; Govi, A.R. (to Dept. of Energy). US Patent 4,076,584. 28 Feb 1978. Filed date 28 Jan 1976. 16p.

PAT-APPL-653,175.

A top mounted nuclear reactor diverse rodged shutdown system utilizing gas fed into a pressure bearing bellows region sealed at the upper extremity to an armature is described. The armature is attached to a neutron absorber assembly by a series of shafts and connecting means. The armature is held in an uppermost position by an electromagnet assembly or by pressurized gas in a second embodiment. Deenergizing the electromagnet assembly, or venting the pressurized gas, causes the armature to fall by the force of gravity, thereby lowering the attached absorber assembly into the reactor core.

- 3790 SELF LOCKING DRIVE SYSTEM FOR ROTATING PLUG OF A NUCLEAR REACTOR. Brubaker, J.E. (to Dept. of Energy). US Patent Application 880,679. 23 Feb 1978. 13p.

The disclosure describes a self-locking drive system for rotating the plugs on the head of a nuclear reactor which is able to restrain plug motion if a seismic event should occur during reactor refueling. A servomotor is engaged via a gear train and a bull gear to the plug. Connected to the gear train is a feedback control system which allows the motor to rotate the plug to predetermined locations for refueling of the reactor. The gear train contains a self locking double enveloping worm gear set. The worm gear set is utilized for its self locking nature to prevent unwanted rotation of the plugs as the result of an earthquake. The double enveloping type is used because its unique contour spreads the load across several teeth providing added strength and allowing the use of a conventional size worm.

- 3791 METHOD FOR REMOVING FISSION PRODUCTS FROM A NUCLEAR REACTOR COOLANT. Colburn, R.P. (to Westinghouse Electric Corp.). US Patent 4,075,060. 21 Feb 1978. Filed date 31 Jul 1974. 6p.

A method is described for removing cesium 137, iodine 131, and tritium from a liquid sodium reactor coolant. Hydrogen is introduced into the liquid sodium causing sodium hydride to precipitate in a cold trap. The cesium, iodine, and tritium isotopes are removed from the liquid sodium by coprecipitating as impurities in the sodium hydride.

- 3792 DISPERSION STRENGTHENED FERRITIC ALLOY FOR USE IN LIQUID-METAL FAST BREEDER REACTORS (LMFBRs). Fischer, J.J. (to International Nickel Co., Inc.). US Patent 4,075,010. 21 Feb 1978. Filed date 5 Feb 1976. 10p.

A dispersion-strengthened ferritic alloy is

provided which has high-temperature strength and is readily fabricable at ambient temperatures, and which is useful as structural elements of liquid-metal fast breeder reactors. 4 tables.

- 3793 NUCLEAR REACTOR WITH LIQUID METAL COOLING. Seel, G.; Hodgson, D.; Grime, C.J. (to Nuclear Power Co. Ltd., London (UK); Deutsches Patentamt, Muenchen (Germany, F.R.)). German(FRG) Patent 2,736,253/A/. 16 Feb 1978. 12p. (In German).

In a sodium cooled fast breeder reactor, the primary vessel is suspended from the ceiling of a safety containment, where the edge of the vessel is made gas-tight in order to contain an inert cover gas. In order to limit uncontrolled movement of the core relative to the control rods, apart from the support construction a second support device is provided for the core.

- 3794 NUCLEAR REACTOR WITH LIQUID METAL COOLING. Seel, G.; Hodgson, D.; Grime, C.J. (to Nuclear Power Co. Ltd., London (UK); Deutsches Patentamt, Muenchen (Germany, F.R.)). German(FRG) Patent 2,736,251/A/. 16 Feb 1978. 13p. (In German).

The reactor tank of a sodium cooled fast breeder reactor is immersed in a safety dome which is open in the upward direction towards the coolant basin and is suspended from the ceiling. A leakage container contains an inert cover gas for the coolant basin and is sealed at the ceiling.

- 3795 NUCLEAR REACTOR WITH LIQUID METAL COOLING. Seel, G.; Hodgson, D.; Grime, C.J. (to Nuclear Power Co. Ltd., London (UK); Deutsches Patentamt, Muenchen (Germany, F.R.)). German(FRG) Patent 2,736,252/A/. 16 Feb 1978. 9p. (In German).

The primary vessel, reactor core and control assemblies are suspended from the ceiling so that they are immersed in sodium in a sodium cooled fast breeder reactor. A collecting vessel is provided around the primary vessel, in order to collect the coolant in the case of a loss of coolant accident.

- 3796 LID CONSTRUCTION FOR NUCLEAR REACTOR VESSEL. Garin, J. (to Westinghouse Electric Corp., East Pittsburgh, PA (USA); Deutsches Patentamt, Muenchen (Germany, F.R.)). German(FRG) Patent 2,729,976/A/. 16 Feb 1978. 29p. (In German).

A rotatable lid for the reactor vessel with closing plates for a sodium cooled reactor is introduced. The eccentric lid plates have annular, metallic and flexible seals, which keep out oxygen, from outside and prevent the escape of radio-active materials in case of containment vessel failure.

- 3797 NUCLEAR REACTOR CORE CATCHING APPARATUS. Barleom, L.; Dorner, S.; Goetzmann, O.; Kussnau, G. (to Gesellschaft fuer Kernforschung). US Patent 4,073,682. 14 Feb 1978. Priority date 21 Dec 1973, German, Federal Republic of (F.R. Germany), 4p.

An apparatus for containing the components of a nuclear reactor core during a core-melt-down accident, comprising a container consisting at least its inner surface of a highly temperature resistant material. 5 claims, 4 figures.

- 3798 REASON ENOUGH FOR THE BREEDER. Wise, C.E. Mach. Des.; 50: No. 3, 26-28, 30-32(9

Feb 1978).

Breeder reactors use 60% of the energy available in natural uranium, while conventional light-water reactors use only 2%. What is more, the vast quantities of uranium tailings, which are presently useless, could be converted by a breeder reactor to plutonium at such a rate as to eliminate the need for mined uranium for 100 years. Proponents of the breeder feel that sufficient momentum has been gained by the project for it to become a reality. The only real threat to breeder technology may come from the laser-enrichment process for uranium now under development by Exxon.

- 3799 BOTTOM COOLER FOR NUCLEAR REACTORS. Friedrich, H.J. (to Interatom, Internationale Atomreaktorbau GmbH). US Patent 4,072,561. 7 Feb 1978. Priority date 11 Aug 1975, German, Federal Republic of (F.R. Germany), 6p.

A cooler for a bottom tray of a liquid metal-cooled nuclear reactor is described, the tray serving as a receptacle for a melting reactor core in the event of a hypothetical accident, comprising coolant distribution and collecting pipes disposed below the bottom tray about the circumference thereof, and a plurality of parallel tubes connecting said distribution and collecting pipes, said tubes being in the shape of involutes running between the center and circumference of said tray. 7 claims, 4 figures.

- 3800 HEAT EXCHANGER WITH INTERMEDIATE EVAPORATING AND CONDENSING FLUID. Fraas, A.P. (to Dept. of Energy). US Patent 4,072,183. 7 Feb 1978. Filed date 29 Nov 1976. 10p. PAT-APPL-745,941.

A shell and tube-type heat exchanger, such as a liquid sodium-operated steam generator for use in nuclear reactors, comprises a shell containing a primary fluid tube bundle, a secondary fluid tube bundle at higher elevation, and an intermediate fluid vaporizing at the surface of the primary fluid tubes and condensing at the surface of the secondary fluid tubes.

- 3801 (ANL-Trans--1121) DEVICE FOR THERMAL SHIELDING OF A REACTOR VESSEL. Defauchy, J.; Malaval, C. Feb 1978. Translation of French Patent Application No. 74 37575 16p. Dep. NTIS, PC A02/MF A01.

The invention deals with a device for thermal shielding of a reactor vessel. More specifically, this invention concerns a thermal shielding device for the vessel of an integrated liquid metal-cooled fast reactor in which the complete primary cooling circuit is located in the interior of the main tank.

- 3802 (ANL-Trans--1119) SYSTEM FOR COOLING THE UPPER WALL OF A NUCLEAR REACTOR VESSEL. Pailla, H.; Schaller, K.; Vidard, M. Feb 1978. Translation of French Patent 7,429,648 17p. Dep. NTIS, PC A02/MF A01.

The invention deals with a cooling system for the upper wall of a nuclear reactor vessel, especially the main tank of a liquid metal-cooled fast reactor.

- 3803 (ANL-Trans--1120) HEAT INSULATING SYSTEM FOR THE HORIZONTAL CLOSING SURFACES FOR A LIQUID METAL COOLED NUCLEAR REACTOR. Jullien, R.; Perons, M.; Venot, R. Feb 1978. Translation of French Patent Application No. 74 30347 12p. Dep. NTIS, PC A02/MF A01.

The invention described concerns a heat

insulating system designed especially for use in a liquid metal cooled nuclear reactor in order to assure thermal shielding of the upper closing surfaces of the vessel of such a reactor, this vessel containing a metal tank confining the volume of liquid cooling metal covered with a neutral gas atmosphere, generally consisting of argon. These closing surfaces in particular carry the closing slab proper consisting of a horizontal concrete wall clad with a stainless steel metal plate on its surface in contact with the neutral gas, rotary plugs, components having the same structure as the slab, occupying one of the openings made in the latter and providing access to the reactor core within the vessel.

- 3804 (ANL-Trans--1124) HEAT INSULATING SYSTEM, ESPECIALLY FOR INSULATING THE UPPER ZONE OF THE ANNULUS SEPARATING THE PRIMARY AND SAFETY VESSELS OF A FAST REACTOR. Lemerrier, G. Feb 1978. Translation of French Patent 2,289,032, May 21, 1976 13p. Dep. NTIS, PC A02/MF A01.

The invention concerns a heat insulating system, more specifically designed for insulating the upper zone of an annulus provided between the primary and safety vessels of a fast reactor, these vessels having a common vertical axis, being open at the top and covered by a horizontal slab, usually of concrete, in which the ends of these vessels or "straps" are seated so as to allow their suspension from the slab.

- 3805 (ANL-Trans--1127) LIQUID-COOLED NUCLEAR REACTOR. Mueller, A. Feb 1978. Translation of German Patent No. 2,013,586 15p. Dep. NTIS, PC A02/MF A01.

The invention concerns a liquid metal cooled nuclear reactor consisting of a stationary reactor vessel containing a fission zone and a stationary component vessel containing heat exchangers and coolant circulating pump, where the component vessel is connected to the reactor vessel by a coaxial pipe in which the coolant flows from the fission zone to the heat exchangers and back.

- 3806 IMPURITY INTERACTION ANALYSIS IN MESH-PACKED COLD TRAPS. Murase, M.; Sumida, I.; Kotani, K.; Yamamoto, H. (Hitachi Ltd., Kawasaki, Japan). Nucl. Technol.; 37: No. 2, 138-147(Feb 1978).

An impurity interaction code for a mesh-packed cold trap was developed, and the properties of the packed trap, such as the trapping efficiency and capacity, were calculated using this code. As a result, the trapping efficiency and capacity are greatly affected by the packing ratio and the wire diameter of the mesh packing. Precipitation of impurity is promoted at the entrance to the mesh packing. This facilitating effect hastens flow blockage and decreases the capacity. Moreover, the capacity decreases with increase in the efficiency.

- 3807 (ANL-Trans--1128) STUDIES NEEDED TO PREVENT THE USE OF EXPANSION BENDS IN LMFBR INTERMEDIATE HEAT EXCHANGERS. Kayser, G. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides). Feb 1978. Translation of French Paper F7/11. Vol. 2: Reactor core coolant circuit, Part F (CONF-750908--9). 11p. Dep. NTIS, PC A02/MF A01.

From 3. international conference on structural mechanics in reactor technology; London, UK (1 Sep 1975).

The LMFBR IHX built in France consists in a vertical tube bundle welded on 2 tube sheets. The secondary sodium flows down a central pipe to the lower collector then up through the tube bundle where it is heated. During the preliminary studies for the PHENIX reactor, it was soon evident that a structure with expansion bends on tubes would lead to prohibitive sheet-thickness. Moreover, the presence of bends increased the total length, made vibration problems more complicated, and, finally made the structure more expensive. Therefore, a solution with straight tubes was studied, and, despite the pressure and diameter increase, the Phenix sheets had the same thickness as Rapsodie's ones. Evidently, the solution of the problems raised by the presence of thermal stresses needed thorough studies and led to the following theoretical and experimental developments: (1) A computer code was written for structural analysis. The structure was divided in annular elements that could be studied by means of the elementary theory of shells and plates; and reduced elastic coefficients were given to the tube sheets to account for the presence of drilled holes. (2) An experimental study was undertaken to determine the reduced elastic coefficients of the tube sheets. (3) A computer code was written to study the primary sodium flow around the tube bundle, and experimental studies were made on a mockup, the fluid being water. (4) The results of the previous code were used to determine, by means of a code for thermal analysis, the temperature field in the bundle both in steady state and transient regimes. Up to now, many transients were performed and the Phenix heat exchangers have been operating quite satisfactorily; this seems to prove the design assumptions were correct.

- 3808 EXPERIMENTAL STUDIES OF THE GROWTH OF AN INTERNALLY HEATED LIQUID POOL IN A SOLID BED. Farhadieh, R.; Baker, L. Jr. (Argonne National Lab., IL). Nucl. Sci. Eng.; 65: No. 2, 394-400(Feb 1978).

Two-dimensional simulation experiments have been conducted on the growth of an internally heated liquid pool in a solid bed. The solid bed is a water-soluble wax, whereas the liquid pool is an electrolyte solution. The liquid pool and the molten solid are mutually miscible. The initial density of the liquid pool is greater than that of the solid at the melting point. The growth characteristics of the hot liquid pool in the solid are affected by the initial density, viscosity, and internal heat generation. The size of the liquid pool is affected only by power density.

- 3809 PROBABILISTIC ANALYSIS OF LIQUID-METAL FAST BREEDER REACTOR ACCIDENT CONSEQUENCES WITH RESPONSE SURFACE TECHNIQUES. Vaurio, J.K.; Mueller, C. (Argonne National Lab., IL). Nucl. Sci. Eng.; 65: No. 2, 401-413(Feb 1978).

Response surface techniques are presented for obtaining the probability distributions of selected consequences of a liquid-metal fast breeder reactor hypothetical core disruptive accident. The uncertainties of the consequences are considered as a variability of the system and model input parameters used in the accident analysis. Probability distributions are assigned to the input parameters, and parameter values are systematically chosen from these distributions. These input parameters are then used in deterministic consequence analyses that are performed by fast-running analogs of the comprehensive mechanistic accident analysis codes. The results of these deterministic consequence analyses are used to generate the coefficients for response surface functions

that approximate the consequences in terms of the selected input parameters. These approximating functions are then used to generate the probability distributions of the consequences with random sampling being used to obtain values for the accident parameters from their distributions. Two different schemes are presented for selecting the knot-point values of the input parameters. The first generates a single second-order polynomial for the entire parameter space; the second generates separate polynomials for specified regions of the parameter space. The calculation of conditional distributions of the consequences and the use of these distributions to define importance distributions of the input parameters are presented. The use of these procedures is illustrated by applications to a postulated loss-of-flow transient with failure to scram in a Clinch River Breeder-type reactor.

- 3810 MODERN APPLICATIONS FOR CAST AND WROUGHT SUPERALLOYS. Green, D.R. (Union Carbide UK Ltd.). Metallurgia; 45: No. 2, 80-85(Feb 1978).

This paper covers casting and wrought alloys for gas turbines, alloys for power generation (nuclear applications), automotive applications, alloys for corrosion resistance, and hot working tools. The section on nuclear applications includes references to the use of a niobium stabilised 20 Cr - 25 Ni Nb austenitic stainless steel for fuel cladding for the A.G.R., and other alloys based on chromium and nickel for such purposes as fuel element components for the P.F.R. and bolts in steam turbine plant.

- 3811 UNIAXIAL COMPRESSIVE STRENGTHS OF CONCRETE FOR TEMPERATURES REACHING 1033 K. Callahan, J.P.; Robinson, G.C. (Oak Ridge National Lab., Tenn. (USA)); Burrow, R.C. (Burns and Roe, Inc., Oradell, N.J. (USA)). Nucl. Eng. Des.; 45: No. 2, 439-448(Feb 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, USA (15 - 19 Aug 1977).

A series of tests was conducted to evaluate effects of sustained heating on the strengths of limestone aggregate concrete for the Clinch River Breeder Reactor Plant. Twenty-four 15.24-cm dia. by 30.48-cm long cylinders were divided into two groups subjected to either open-hot or closed-cold conditions. All specimens were heated for 14 days at soak temperatures of 450, 644, 839 and 1033 K. Unconfined compressive strengths were determined at these temperatures for the open-hot cylinders and after cooling for the closed-cold set. The results provided realistic upper and lower bounds for published strength data.

- 3812 (EURFNR--1427) MIPLO: A GRAPHIC PROGRAM FOR THE REPRESENTATION OF TEMPERATURE DISTRIBUTIONS IN BUNDLES WITH HEXAGONAL ROD CONFIGURATION USED IN CONNECTION WITH MISTRAL-II. Frey, H.H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorbauelemente). Feb 1978. Translation by T.J. Clark of KFK--2393 51p. Dep. NTIS, PC A04/MF A01.

Work performed under United States--Euratom Fast Reactor Exchange Program.

A digital FORTRAN-IV program is described, which gives the temperature distribution in hexagonal LMFBR rod bundles. For bundles with 7 to 217 rods, the following variations for temperature profiles for channel-center and surface temperatures are possible: (a) wall-traverse, (b) corner-traverse, (c) circumference (peripheral channels), (d) axial

temperature curve of individual sub-channels, (e) axial development of profile, and (f) desired route. In variants a, b, c, and f, the temperature profiles of the last computer planes in the bundle exit are plotted. The representations can be inscribed according to the variant, and provided with explanations, so that a drawing with sufficient information will be produced. The report contains the description of the read-statements in Subroutine M3PLOT and shows a selection of examples of its applications.

- 3813 MECHANICAL RESPONSE OF LMFBR AND P1 CLADDING TUBES UNDER TRANSIENT HEATING. Youngdahl, C.A.; Ariman, T.; Pierce, R.D.; Lepacek, B.E. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 45: No. 2, 383-391(Feb 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, USA (15 - 19 Aug 1977).

Type 316 stainless steel tubing specimens comparable to LMFBR cladding were burst tested with relatively constant internal pressure in the 219-836 psi range and with increasing temperature. Continuous measurements of diameter change, temperature, and pressure were recorded as the samples were heated to temperatures near the melting point at rates from 10-1800°F/sec. The effects of varying initial wall thickness, cold work level, length, and thermal experience were explored. Ductile failures were observed at 10°F/sec, and stable strains at time of failure were greater than those reported by HEDL. At 200°F/sec the initially 20% cold-worked, 15-mil wall tubing produced brittle failures; while initially 40% cold worked, 10-mil wall samples displayed a mixture of ductile and brittle features. At approximately 1000°F/sec the behavior of the latter material was predominately brittle, although stable strains as large as 6% were observed. Failure temperatures were generally above 2000°F.

- 3814 CONDUCTION CENTRIFUGAL ELECTROMAGNETIC PUMP. Bonciani, L.; Grassi, W. (Pisa Univ. (Italy)). Energ. Nucl. (Milan); 25: No. 2, 92-102(Feb 1978).

After a general introduction on the characteristics of conduction and induction electromagnetic pumps, the model of a particular type of conduction centrifugal electromagnetic pump is at first illustrated; a simplified theory of the pump operational principle is then given, as well as a relation between pump head and capacity. Working at null capacity is also described from both the theoretical and experimental viewpoints.

- 3815 HEAT TRANSFER AROUND A CIRCULAR CYLINDER IN LIQUID SODIUM CROSSFLOW. Ishiguro, R.; Kumada, T.; Sugiyama, K. (Hokkaido Univ., Sapporo (Japan). Faculty of Engineering). Hokkaido Daigaku Kogakubu Kenkyu Hokoku; No. 86, 133-147(Feb 1978). (In Japanese).

The local and average heat transfer coefficients are experimentally determined for sodium flow normal to a circular cylinder. A sodium loop with special specifications is designed for the purpose. The measurement covers the range of Reynolds number 360 -- 16,900 with three test cylinders of different diameter. The data are compared with the analytical values obtained by Grosh and Cess under the assumption of inviscid flow. Their analysis predicts satisfactorily heat transfer coefficients at the front part of a cylinder where a boundary layer is developing. However, the experimental data are much smaller than the

analytical values over the region beyond separation point. It is concluded that the assumption of inviscid flow leads to large errors in the prediction of both local and average heat transfer coefficients even in such a fluid of low Prandtl number as sodium. A numerical analysis is also tried under the assumption of viscous flow at a Peclet number of 7.8 and is in agreement with the experimental data in reasonable accuracy.

- 3816 (JAPFNR--318) PARAMETER SURVEY ON THE SATURATION BOILING PROCESS FOR CREVICE SODIUM REMOVAL. Kamei, M.; Kanbe, M.; Nakamura, K.; Fukada, T.; Ishii, Y.; Saito, R.; Mimoto, Y. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Feb 1978. Translation of PNC-N--941-78-49 22p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

It was found from the experiences that sodium removal from small crevices by conventional methods was difficult. And, in order to improve the operation, a saturation boiling process was applied and experiments using test pieces have started. Since the purpose of this process is to remove final remaining deposits, bulk sodium removal must be done before. The saturation boiling process is a modification of the water charging process. When water is charged into the process cell or tank, the pressure inside is adjusted so that boiling of the water is initiated. The formation of hydro-oxide membranes at the surface of the sodium prevents further reaction between sodium and water. Results have been obtained from test pieces of bolts and crevices of flat plates and cylinders.

- 3817 (JAPFNR--319) EXPERIENCE ON SODIUM REMOVAL FROM VARIOUS COMPONENTS. Kamei, M.; Kanbe, M.; Yagisawa, H.; Saasaki, S.; Kataoka, H.; Fukada, T.; Ishii, Y.; Saito, R.; Mimoto, Y. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Feb 1978. Translation of PNC-N--941-78-48 39p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

Descriptions are presented for studies on the following methods for removal of sodium from the components of sodium plants: steam cleaning for the 50 MW Steam Generator, secondary proto-type pump of "JOYO" and Dummy fuel assembly of "JOYO"; alcohol cleaning for the Sector Model of the Intermediate Heat Exchanger (IHX) of "JOYO", a sector model of Sodium-to-Air cooler of "JOYO" and a proto-type isolation valve of "JOYO"; and cleaning by vacuumization at high temperature for the Regenerative Heat Exchanger. This report describes the outline of the Sodium Disposal Facility and experience of sodium removal processing on the 50 MW Steam Generator, the crevices of the experimental subassemblies, the Fuel Handling Machine of "MONJU", and the regenerative heat exchanger of the Sodium Flow Test Facility.

- 3818 SODIUM CORROSION OF WEAR RESISTANT COBALT ALLOYS (STELLITE). Borgstedt, H.U. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). Werkst. Korros.; 29: No. 2, 81-91(Feb 1978). (In German).

STELLITE 6B and 6H suffer changes of the morphology and composition of their surfaces during the exposure to flowing sodium in a loop constructed of austenitic stainless steels. At high temperatures (973 K) mass transfer causes

the formation of a metallic phase as an unequal cover of the surfaces. At lower temperature (873 K and less) a chromium rich oxide layer is formed. The metallic particles exceed the materials in an unfavorable manner. Losses of material do not play an important role in the sodium corrosion of the group of materials.

- 3819 ACOUSTIC MEASUREMENTS OF THE BOILING STABILITY TESTS ON THORS SODIUM LOOP. Sheen, S.H.; Bobis, J.P.; Carey, W.M. (Argonne National Lab., IL). IEEE Trans. Nucl. Sci.; NS-25: No. 1, 255-258(Feb 1978).

Acoustic data of boiling stability tests on the THORS (Thermal-Hydraulic Out-of-Reactor Safety) facility were obtained using three sodium-immersible high temperature microphones. The data was analyzed in both the time and frequency domains and provides the following information: (1) the acoustic signal due to sodium boiling was clearly observed; (2) the signal level and the repetition rate of boiling pulses are directly proportional to the applied heat flux; (3) a typical boiling pulse consists of a high frequency signal due mainly to the bubble collapses and a low frequency void oscillation; (4) the frequency spectra of the boiling and background pulses can be mostly assigned to various acoustic resonance frequencies of the THORS loop. The characterization of these data is pertinent to the design of sodium boiling detection systems.

- 3820 FINITE ELEMENT ELASTIC-PLASTIC ANALYSIS OF LMFBR COMPONENTS. Levy, A.; Pifko, A.; Armen, H. Jr. (Grumman Aircraft Engineering Corp., Bethpage, N.Y. (USA). Research Dept.). Nucl. Eng. Des.; 45: No. 2, 411-418(Feb 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, USA (15 - 19 Aug 1977).

The present effort involves the development of computationally efficient finite element methods for accurately predicting the isothermal elastic-plastic three-dimensional response of thick and thin shell structures subjected to mechanical and thermal loads. This work will be used as the basis for further development of analytical tools to be used to verify the structural integrity of liquid metal fast breeder reactor (LMFBR) components. The methods presented here have been implemented into the three-dimensional solid element module (HEX) of the Grumman PLANS finite element program. These methods include the use of optimal stress points as well as a variable number of stress points within an element. This allows monitoring the stress history at many points within an element and hence provides an accurate representation of the elastic-plastic boundary using a minimum number of degrees of freedom. Also included is an improved thermal stress analysis capability in which the temperature variation and corresponding thermal strain variation are represented by the same functional form as the displacement variation. Various problems are used to demonstrate these improved capabilities.

- 3821 ANALYSIS OF DYNAMICALLY LOADED NON-LINEAR STRUCTURES. Davidson, I. (9 Dale Lane, Appleton, Warrington, UK); Bradbury, J.N. (Nuclear Power Co. (Risley) Ltd. (UK)). Nucl. Eng. Des.; 45: No. 2, 399-410(Feb 1978).

From 4. international conference on structural mechanics in reactor technology; San Francisco, USA (15 - 19 Aug 1977).

Small models of a proposed prestressed concrete containment structure for a sodium cooled fast breeder reactor have been made and tested. They were partly filled with water and

loaded internally by detonating explosive charges. Prior to the tests the model was analysed by an axisymmetric dynamic relaxation computer program. No difficulties occurred with the analyses which were arranged to print out, *inter alia*, stresses, deflections and cracks at suitable time intervals during and after the pressure excursions which lasted for up to 3000 μ sec. A previous paper describing the program, has shown how cracks can be calculated simply and accurately. The correlation of computer and test results is discussed.

- 3822 MATERIALS REQUIREMENTS FOR LIQUID METAL FAST BREEDER REACTORS. Bennett, J.W.; Horton, K.E. (Energy Research and Development Administration, Washington, DC). Metall. Trans., A; 9: No. 2, 143-149 (Feb 1978).

Materials requirements for Liquid Metal Fast Breeder Reactors (LMFBRs) are quite varied with requisite applications ranging from ex-reactor components such as piping, pumps, steam generators and heat exchangers to in-reactor components such as heavy section reactor vessels, core structurals, fuel pin cladding and subassembly flow ducts. Requirements for ex-reactor component materials include: good high temperature tensile, creep and fatigue properties; compatibility with high temperature flowing sodium; resistance to wear, stress corrosion cracking, and crack propagation; and good weldability. Requirements for in-reactor components include most of those cited above for ex-reactor components as supplemented by the following: resistance to radiation embrittlement, swelling and radiation enhanced creep; good neutronics; compatibility with fuel and fission product materials; and resistance to mass transfer via flowing sodium. Extensive programs are currently in place in a number of national laboratories and industrial contractors to address the materials requirements for LMFBRs. These programs are focused on meeting the near term requirements of early LMFBRs such as the Fast Flux Test Facility and the Clinch River Breeder Reactor as well as the longer term requirements of larger near-commercial and fully-commercial reactors.

- 3823 (JAPFNR--398) SODIUM VAPOR DEPOSITION ON A SIMULATED MODEL OF AN LMFBR ROTATING SHIELD PLUG ANNULUS (2). Himeno, Y. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). Feb 1978. Translation of PNC-N-941 78-44 27p. Dep. NTIS (US Sales Only), PC A03/MF A01.

The paper discusses sodium vapor deposition on a simulated LMFBR rotating shield plug annular walls, both experimentally and on the basis of numerical calculation. Experiments were carried out to investigate the effects of cover gas natural convection within the annulus to local and average deposition rates. It was found that the generalized average deposition rate, which is similar to Sherwood number, depends on Rayleigh number. Numerical calculation of saturated vapor condensation rate along the wall was also carried out to demonstrate the effect of vapor condensation to be small to real deposition phenomena. The results suggested that deposition process is controlled by the mist transfer and their deposition onto the wall.

- 3824 (JAPFNR--405) SODIUM VAPOR DEPOSITION ON A SIMULATED MODEL OF AN LMFBR ROTATING SHIELD PLUG ANNULUS (1). Himeno, Y. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering

Center). Feb 1978. Translation of PNC-N-941 78-43 23p. Dep. NTIS (US Sales Only), PC A02 MF A01.

The paper presents sodium vapor deposition rates onto annular walls of the test assemblies, which consist of three annuli and are mounted in the cover gas space of the sodium vessel. Experiments carried out include measurements of deposition rates and their dependence on pool temperature, cover gas pressure, and temperature drop across cover gas space above the pool. The results are discussed and compared with those derived from previously known theories on evaporation of vapor in the presence of mist formation. The experimental results and theoretical predictions did not agree. The dependence of the experimental data were known to be caused by the evaporation rate dependence on these factors.

- 3825 FUEL FAILURE DETECTION EXPERIMENT USING UO_2 -LOADED SODIUM IN-PILE LOOP. Sakai, E.; Fujino, M.; Imai, K.; Yoshida, H.; Katagiri, M.; Terada, H.; Itoh, H. (Jpn At Energy Res Inst, Tokaimura, Ibarakiken). IEEE Trans. Nucl. Sci.; NS: No. 1, 266-277 (Feb 1978). (CONF-771023--).

From Nuclear science symposium; San Francisco, CA, USA (19 Oct 1977).

Performance of various types of fuel failure detection systems has been studied using a uranium oxide loaded sodium in-pile loop. The detection systems include a delayed neutron monitor (graphite or polyethylene moderator), a moving-wire precipitator (detecting beta- and gamma-rays separately), a cover gas gamma-ray spectrometer (single Ge(Li) or an anticoincidence hyperpure Ge), and a Si(Li)-precipitator. Construction and performance of these detection systems are described as well as the methods of the detection efficiency determination. A method is introduced which gives the fission product release fraction from the fuels and the fission product transfer rate from sodium to cover gas, using only the fission product concentrations in the cover gas. 15 refs.

- 3826 ACOUSTICS OF THE SODIUM LOOP SAFETY FACILITY P2 EXPERIMENT. Anderson, T.T.; Carey, W.M.; Oats, M. (Argonne Natl Lab, Ill). IEEE Trans. Nucl. Sci.; NS25: No. 1, 112-115 (Feb 1978). (CONF-771023--).

From Nuclear science symposium; San Francisco, CA, USA (19 Oct 1977).

In the highly instrumented P2 experiment to simulate an unprotected loss of flow in an LMFBR core, two high-temperature sodium-immersed microphones were used to obtain the acoustic scenario of the 9-second flow coastdown, boiling sodium coolant, fuel failure, and re-establishment of flow after reactor scram. Acoustic data from this and previous experiments indicate that singular larger voids are formed in the sodium boiling process. These singular larger voids were observed to collapse upon contacting the subcooled sodium or structures, at repetition rates comparable to formation rates of sodium vapor bubbles. The acoustic observations were made in the presence of sodium exposed to argon cover gas at LMFBR operating conditions, and in the presence of fission gas from fuel pin failure. 6 refs.

- 3827 EFFECT OF NP-239 ON FAST REACTOR BURNUP CALCULATION. Takeda, S.; Miki, K. (Hitachi Ltd., Kawasaki, Kanagawa (Japan). Atomic Energy Research Lab.). J. Nucl. Sci. Technol. (Tokyo); 15: No. 2, 156-158 (Feb 1978).

BREEDING RATIO; BURNUP; CROSS SECTIONS; LMFBR

=====

TYPE REACTORS; NEPTUNIUM 239; REACTIVITY; REACTOR OPERATION

- 3828 COULD NUCLEAR WASTE RECYCLED IN REACTORS. Hermann, R. Arch. Energiewirtsch.; 32: No. 2, 119-123 (Feb 1978). (In German).

To reduce the danger due to spent nuclear fuel, it is best to use this radioactive waste in new fuel elements and thus to recycle it in the reactor. The fundamentals of a possible technology are discussed.

- 3829 TRANSIENT BOILING OF SODIUM IN A SINGLE-PIN GEOMETRY UNDER LOSS-OF-FLOW CONDITIONS. Kikuchi, Y.; Haga, K. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). J. Nucl. Sci. Technol. (Tokyo); 15: No. 2, 100-108 (Feb 1978).

Transient sodium boiling experiments have been conducted in an electrically heated single-pin annular channel under loss-of-flow conditions. Each run was made by reducing or stopping the flow at constant heater power. Records were obtained of the changes of temperatures, flow rates, pressures and voids. The experimental results were compared with analytical calculations. There was no strong effect of temperature ramp rate on the incipient-boiling (IB) wall superheat in the range of the present experimental conditions. The pressure rise at the initial vaporization compared well with the vapor pressure corresponding to the IB wall superheat. The pressure pulse occurred at the vapor collapse correlated reasonably well with the collision velocity between the re-entrant liquid columns, but the measured value was less than the theoretical value predicted from the sodium hammer analysis. The observed voiding pattern agreed fairly well with the calculation by the NAIS-P2 code (single-bubble slug expulsion model). The initial expulsion acceleration of liquid agreed well with the theoretical prediction from the one-dimensional liquid column expulsion analysis. The thickness (0.05 and 0.29 mm) of the residual liquid film became thinner with higher IB wall superheat.

- 3830 WORLD'S POWER DEVELOPMENT AND IMPORTANCE OF NUCLEAR POWER. Neumann, J. (Ceskoslovenska Komise pro Atomovou Energii, Prague). Energetika (Prague); 28: No. 2, 81-86 (Feb 1978). (In Czech).

Based on data on the current state of world power submitted to the 9th World Power Conference in 1974, factors are discussed affecting the future development of power generation in the world. Particular attention is devoted to nuclear power, mainly with respect to Czechoslovak needs. Reactors of the WWER-440 and WWER-1000 type are briefly described and questions are discussed of the reactor fuel cycle, safety, and process heat utilization. The BN-600 fast reactor is also briefly described.

- 3831 ACCIDENT ALARM IN STEAM GENERATORS IN SODIUM COOLED FAST REACTOR POWER PLANTS. II. MEASURING THE ACOUSTIC SPECTRUM INDUCED BY THE SODIUM/WATER REACTION AS A METHOD OF BREAKDOWN DETECTION OF LIQUID SODIUM-HEATED STEAM GENERATORS. Matál, O.; Martoch, J. (Vyzkumny Ústav Zavodu Energetického Strojírenství, Brno-Slatina (Czechoslovakia)); Taraba, O.; Hanke, V. (Česke Vysoké Učení Technické, Prague (Czechoslovakia)). Jad. Energ.; 24: No. 2, 41-44 (Feb 1978). (In Czech).

Conditions were simulated in the economizer of a steam generator of water leaks in sodium

at a sodium flow of 0.62×10^{-3} to 1.24×10^{-3} m³/s and a sodium temperature of 320 to 380 degC by injecting water at a pressure of 6 to 10 MPa which roughly corresponds to conditions in an economizer of an actual steam generator with leaks within the limits of 0.01 to 0.3 g/s. The leak was recorded by acoustic detectors at all observed sodium flow rates and temperatures. The mean signal-to-noise ratio was in all cases greater than 2. At the assumed 25 dB noise level of the real steam generator of micromodular design it may be assumed that using existing acoustic detectors with waveguides a 0.02 g/s leak of water into sodium may be detected. The measurements showed that the technical standard of the equipment is at least as good as that of the flowmeter system of accident monitoring.

- 3832 NATURAL EVAPORATION OF SODIUM UNDER THE CONDITIONS OF MIST FORMATION, (2). DISCUSSIONS OF CRITICAL SUPERSATURATION MODEL BY OBSERVATION AND PHOTOGRAPHING OF MIST REGION. Kumada, T.; Ishiguro, R. (Hokkaido Univ., Sapporo (Japan). Faculty of Engineering); Shimizu, T. Nippon Kikai Gakkai Ronbunshu; 44: No. 378, 608-616 (Feb 1978). (In Japanese).

The mist generated when sodium evaporates naturally in the atmosphere of argon kept at room temperature was observed with helium-neon gas laser. The dust free range is formed between the region of mist generation and liquid surface mainly by the effect of heat migration. The inside border of the region of mist generation is separated from the liquid surface as the temperature at sodium surface rises. The supersaturation on the inside border of the region of mist generation is extremely large. The mist visible to naked eye is generated when the temperature of the sodium liquid surface rises to around 270 deg. C. As a result, two assumptions of the critical supersaturation model were evaluated when the temperature of evaporating surface was sufficiently higher than that of gas phase and the amount of evaporation was comparatively small.

- 3833 OUTLINE OF SODIUM-WATER REACTION TEST IN CASE OF LARGE LEAK WITH SWAT-3 TESTING EQUIPMENTS. Sato, M. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). Doryokuro Giho; No. 25, 54-67 (Feb 1978). (In Japanese).

The key component in sodium-cooled fast reactors is steam generators, and the sodium-water reaction owing to the break of heating tubes may cause serious damages in equipments and pipings. The main factor controlling this phenomenon is the rate of leak of water. When the rate of water leak is small, the propagation of heating tube breaking may occur owing to 'wastage phenomenon', on the other hand, when the rate of water leak is large, the phenomena of explosive pressure and flow occur due to the reaction heat and a large quantity of hydrogen generated by the reaction. In PNC, the testing equipments of SWAT-2 for small water leak and SWAT-1 for large leak were constructed, and the development test has been carried out to establish the method of safety design experimentally. The synthetic test equipment for the safety of steam generators, SWAT-3, was constructed to carry out the large water leak test in the scale close to actual plants. The object of the test, the outline of the test equipment, the phenomena of pressure and flow in the water injection test, the confirmation of the occurrence of secondary breaking of adjacent heating tubes, and the disposal of reaction products are described in this paper. This test is still going on, and the

final conclusion will be reported later.

- 3834 NUCLEAR REACTOR APPARATUS. Wade, E.E. (to Westinghouse Electric Corp.). US Patent 4,071,402. 31 Jan 1978. Filed date 9 Aug 1976. 6p.

A lifting, rotating and sealing apparatus for nuclear reactors utilizing rotating plugs above the nuclear reactor core is described. This apparatus permits rotation of the plugs to provide under the plug refueling of a nuclear core. It also provides a means by which positive top core holddown can be utilized. Both of these operations are accomplished by means of the apparatus lifting the top core holddown structure off the nuclear core while stationary, and maintaining this structure in its elevated position during plug rotation. During both of these operations, the interface between the rotating member and its supporting member is sealingly maintained.

- 3835 MEASURING FLUID PRESSURE. Lee, A.S. (to English Electric Co. Ltd.). British Patent 1,499,100/A/. 25 Jan 1978. 5p.

A method and apparatus are described for measuring the pressure of a fluid having characteristics that make it unsuitable for connection directly to a pressure gauge. The method is particularly suitable for the periodic measurement of the pressure of a supply of liquid Na to Na-lubricated bearings of pumps for pumping Na from a reservoir to the bearing via a filter, the reservoir being contained in a closed vessel containing an inert blanket gas, such as Ar, above the Na.

- 3836 NUCLEAR REACTOR REMOVABLE RADIAL SHIELDING ASSEMBLY HAVING A SELF-BOWING FEATURE. Pennell, W.E.; Kalinowski, J.E.; Waldby, R.N.; Rylatt, J.A.; Swenson, D.V. (to Dept. of Energy). US Patent 4,070,241. 24 Jan 1978. Filed date 16 Feb 1977. 6p. PAT-APPL-769,371.

A removable radial shielding assembly for use in the periphery of the core of a liquid-metal-cooled fast-breeder reactor, for closing interassembly gaps in the reactor core assembly load plane prior to reactor criticality and power operation to prevent positive reactivity insertion is described. The assembly has a lower nozzle portion for inserting into the core support and a flexible heat sensitive bimetallic central spine surrounded by blocks of shielding material. At refueling temperature and below the spine is relaxed and in a vertical position so that the tolerances permitted by the interassembly gaps allow removal and replacement of the various reactor core assemblies. During an increase in reactor temperature from refueling to hot standby, the bimetallic spine expands, bowing the assembly toward the core center line, exerting a radially inward gap-closing-force on the above core load plane of the reactor core assembly, closing load plane interassembly gaps throughout the core prior to start-up and preventing positive reactivity insertion.

- 3837 NUCLEAR REACTOR FUEL TRANSFER SYSTEM. Wade, E.E. (to Westinghouse Electric Corp.). US Patent 4,069,099. 17 Jan 1978. Filed date 23 Feb 1976. 8p.

A system for transferring fuel assemblies between a nuclear reactor core and a fuel transfer area while the fuel assemblies remain completely submerged in a continuous body of coolant is described. The system comprises an in-vessel transfer machine and a valve capable of isolating the reactor vessel from the

remainder of the fuel transfer system. The in-vessel transfer machine consists of a positioning mechanism and a lifting mechanism. The positioning mechanism is composed of a plurality of rotatable plugs located on the reactor vessel head. The lifting mechanism which is capable of lifting or lowering a chosen fuel assembly is disposed on a tilted plug at an angle relative to the tilted plug's plane of rotation. The rotatable plugs together with the lifting mechanism are capable of transferring a fuel assembly between the fuel transfer area and any core location or between any two core locations while the fuel assembly remains completely submerged in a continuous body of coolant.

- 3838 EX-VESSEL NUCLEAR FUEL TRANSFER SYSTEM. Wade, E.E. (to Westinghouse Electric Corp.). US Patent 4,069,098. 17 Jan 1978. Filed date 23 Feb 1976. 8p.

A system for transferring fuel assemblies between a fuel transfer area and a fuel storage area while the fuel assemblies remain completely submerged in a continuous body of coolant is described. A fuel transfer area filled with reactor coolant communicating with the reactor vessel below the reactor coolant level provides a transfer area for fuel assemblies in transit to and from the reactor vessel. A positioning mechanism comprising at least one rotatable plug disposed on a fuel transfer tank located outside the reactor vessel cooperates with either the fuel transfer area or the fuel storage area to position a fuel assembly in transit. When in position, a transporting mechanism cooperating with the positioning mechanism lifts or lowers a chosen fuel assembly. The transporting mechanism together with the positioning mechanism are capable of transferring a fuel assembly between the fuel transfer area and the fuel storage area.

- 3839 ADSORPTION SEAL FOR NUCLEAR REACTORS. Cooper, M.H. (to Westinghouse Electric Corp.). US Patent 4,069,100. 17 Jan 1978. Filed date 13 Feb 1975. 6p.

A system is disclosed for reducing the possibility of leakage of fission gases from a nuclear reactor of the gas-cooled type or of a type utilizing a liquid coolant, for example a liquid metal with a cover gas above free surfaces of the coolant. The disclosed system reduces the leakage of fission gases from the cover gas through joints between a pressure vessel and a pressure vessel head in the nuclear reactor by installing an adsorbent material between two seals placed in the joints. This adsorbent material provides sufficient delay time to allow all radioactive isotopes, except long-lived nuclides, to decay to innocuous concentrations before escaping from the pressure vessel. 5 claims, 4 figures.

- 3840 (EURFNR--1413) EXPERIMENTAL DETERMINATION OF TEMPERATURE FIELDS IN SODIUM-COOLED ROD BUNDLES WITH HEXAGONAL ROD ARRANGEMENT AND GRID SPACERS. Moeller, R.; Tschoeke, H.; Kolodziej, M. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). Inst. fuer Reaktorbauelemente. 2 Jan 1978. Translation by T.J. Clark of KFK--2356, Jan 1977. 87p. Dep. NTIS, PC A05/MF A01.

Work performed under United States--Euratom Fast Reactor Exchange Program.

The development of valid methods of calculation for the determination of local three-dimensional temperature fields in rod bundle geometries is studied. In order to get

more precise short-term and mid-term design documents, among others, the azimuthal temperature distributions were determined as a function of the heated length in the cladding tubes of the "critical" rods in the wall region with the inclusion of grid-shaped spacers. This is the principal subject of this report. Parallel to these experiments, the temperature profiles in bundle cross-sections were also measured. Since, for the calculation of local temperature distributions in the cladding tubes of rod bundles, the precise knowledge of the average sub-channel temperatures is a necessary prerequisite, two characteristic experiments (uniform heating and oblique load) for measured sub-channel temperatures are reported on.

- 3841 POWER/ENERGY PROGRESS REPORT: A PASTICHE OF DEVELOPMENTS PROMISES NEAR AND LONG-TERM APPLICATIONS. Kaplan, G. IEEE Spectrum; 15: No. 1, 49-53 (Jan 1978).

A progress report on current energy and power projects covers developments in power generation and transmission, improvements in hardware, techniques for dry-cooling, computerized load management, and breeder safety. A new power station in Prudhoe Bay, Alaska, is farther north than any electric system in the world. Other generating advances include generator/motors for pumped storage, an innovative static excitation system, turbines to prevent subsynchronous resonance, superconductivity-based generators, and solid-state converters. Research on ultra-high voltage (UHV) and high-voltage direct current (HVDC) transmission has progressed along with improvements in relays and gas-insulated switch gear. Relief for power plant siting problems may be available through dry-cooling techniques. Research on computer software is aimed at better control of loads and more-efficient and economic use of equipment. British safety tests on an experimental liquid metal fast breeder reactor are encouraging. Other electric power energy-related projects are listed, including the total estimated funding and the funding agencies (where available).

- 3842 FLEX: A CODE FOR CALCULATING NEUTRON FLUX MICRODISTRIBUTIONS IN LATTICES CONSISTING OF HEXAGONAL FUEL ASSEMBLIES. Agthe, G.; Mai, W.; Moeller, W.; Mueller, A. (VEB Kombinat Kraftwerksanlagenbau, Berlin (German Democratic Republic)). Kernenergie; 21: No. 1, 18-22 (Jan 1978). (In German).

Physical and mathematical fundamentals as well as the coarse structure of the FLEX code are reviewed. In a flexible manner this programme allows one to calculate two-dimensional neutron flux and power density microdistributions in hexagonal geometry for both regular lattices of fuel elements as they exist in zero power reactors and lattices consisting of hexagonal fuel assemblies of WWER type or FBR type reactors. The gaps between fuel rod bundles provided for reasons of design (assembly cans, coolant) can be taken into account in detail. Some results of recalculating test set-up experiments are given.

- 3843 RADIATION BEHAVIOR OF VITREOUS SILICA. Primak, W. (Argonne National Lab., IL). Nucl. Sci. Eng.; 65: No. 1, 141-196 (Jan 1978).

Several properties of vitreous silica relevant to application as an optical element in a reactor were studied, including density, refractive index, optical path, stress relaxation, and optical absorption. The

irradiations were in fields characteristic of the fuel regions of several operating reactors and at temperatures up to 370°C. The radiation-induced changes in optical path are significant, but are much smaller than those in the density and refractive index because the last two are of opposite sign. The thermal coefficient of optical path causes changes of comparable magnitude. Optical absorption is not a serious matter in the upper end of the visible region.

- 3844 EXPERIMENTAL INVESTIGATION OF NEUTRONS STREAMING THROUGH THE GRID-PLATE SHIELD OF LIQUID-METAL FAST BREEDER REACTORS. Uchida, S. (Hitachi Ltd., Igaraki, Japan); Kitamura, M.; Miyasaka, S. Nucl. Sci. Eng.; 65: No. 1, 155-159 (Jan 1978).

Experiments were undertaken to determine the distribution of neutron flux on the core support grid plate of liquid-metal fast breeder reactors by using a semimockup of the grid-plate shield consisting of carbon-steel and aluminum slabs. The experiments were calculated with a conventional calculational procedure containing the TRD-3 two-dimensional removal diffusion code. It was demonstrated that the calculated radial distribution of the fast neutron reaction rates of $^{151}\text{In}(n,n')$ /sup 115m/In agreed with the measured values within a factor of 2.

- 3845 VOID DISTRIBUTION IN BOILING POOLS WITH INTERNAL HEAT GENERATION. Kazimi, M.S. (Massachusetts Inst. of Tech., Cambridge); Chen, J.C. Nucl. Sci. Eng.; 65: No. 1, 17-27 (Jan 1978).

An analytical and experimental investigation of void distribution in internally heated boiling pools is presented. The experimental investigation involved measurement of the vertical profile of the void fraction and the average void fraction in an aqueous solution of ZnSO_4 . If the results of this investigation are extended to a boiling pool of UO_2 , it is predicted that substantial voiding will occur even when only a very small fraction of nominal operating power density in nuclear reactors is consumed in internal vaporization.

- 3846 SPACE--TIME DYNAMICS OF A FAST BREEDER REACTOR FOR LOCALIZED DISTURBANCES. Doshi, J.B.; Grossman, L.M. (Univ. of California, Berkeley). Nucl. Sci. Eng.; 65: No. 1, 106-129 (Jan 1978).

A method of analysis is developed for nuclear reactor accident initiating events that are localized in space. The method is based on a flux factorization technique, accounting for the flux shape changes taking place near the region of perturbation. In the steady state, the neutron shape functions are expanded in a series of eigenfunctions of the steady-state group removal operator. During the unsteady state, the time-dependent group shape functions are expanded in a series of the same stationary eigenfunctions with time-dependent Fourier coefficients. An auxiliary function is added to this expansion to take account of the spatial variation of the spectral hardening of neutrons in the immediate vicinity of the disturbed region. From the resulting representation of the group shape functions, the equations to be satisfied by the time-dependent Fourier coefficients and the time-dependent auxiliary shape function due to the disturbed region are developed consistently. A typical large [1000-MW(e)] liquid-metal fast breeder reactor with two radial core zones of different enrichments is analyzed by the above method. The transient initiating perturbation is taken to be a

specified rate of coolant voiding from a single subassembly in the reactor core. The results show a strong dependence of the reactivity added on the radial location of the voiding perturbation and on the rate of voiding.

- 3847 **FAILED FUEL PIN PERFORMANCE IN FAST METAL-COOLED REACTORS.** Plitz, H. (Gesellschaft fuer Kernforschung mbH, Karlsruhe, Ger.). Nucl. Technol.; 37: No. 1, 48-58 (Jan 1978).

Experience with continued operation of failed mixed-oxide fuel pins in liquid-metal-cooled reactors or in-pile sodium loops is available from a variety of beyond-fuel-failure experiments. The phenomena and effects on a large reactor system of continued beyond-fuel-failure operation are not well understood, but, except for the release and deposition of fission products and the chemical reaction of sodium coolant to oxide fuel, leading to pin swelling, no failure propagation due to continued operation of failed fuel pins has been observed. For economic reactor operation, further investigations are needed to establish a catalog of fuel failure types, sizes, and locations to describe the time-dependent effects of continued operation on reactor operation, shutdown requirements, instrumentation, surveillance, circuit systems, contamination, maintenance systems, and plant efficiency.

- 3848 **ANALYSIS OF THE CENTRAL BLOCKAGE WAKE IN AN LMFBR-SUBASSEMBLY.** Arai, M.; Hirata, N. (Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan). Research and Development Center). Nucl. Eng. Des.; 45: No. 1, 127-132 (Jan 1978).

A simple model is presented to estimate the wake size and the average velocity of the circulating flow downstream of a central blockage in a subassembly. A computer code, newly developed for the purpose was used to examine the model results. It is shown that both the model results and the code closely fit experimental results. It is seen that the size of a turbulent wake is nearly independent of inlet coolant velocity, whereas the velocity of the circulating flow in the wake is nearly proportional to the inlet velocity. Both the length of the wake and the velocity in it increase as the blockage increases. The maximum temperature is reached with a blockage of more than 0.6 of the subassembly cross section, and varies inversely with inlet velocity.

- 3849 **SIMPLE FORCED DIVERSION MODEL FOR STUDY OF THERMAL-HYDRAULIC TRANSIENTS IN LMFBR SUBASSEMBLY.** Chen, W.L.; Grohms, M.A.; Ishii, M. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 45: No. 1, 53-66 (Jan 1978).

The generalized simple, transient, integral energy balances based on the average properties for the fuel and cladding have been used in a new multichannel thermal-hydraulic model for calculating the transient behavior of coolant in the rod bundle. This model was developed to provide a simple useful tool for analyzing the flow and thermal transients in a rod bundle with reasonable accuracy, and to understand the fundamental characteristics of flow in the rod bundle under both normal and abnormal condition of reactor-core operation.

- 3850 **ENERGY FROM THE ATOM.** Walton, G.N. (Imperial Coll. of Science and Technology, London (UK). Dept. of Chemical Engineering and Chemical Technology). Educ. Chem.; 15: No. 1, 18-21 (Jan 1978).

A brief account is given of the source of nuclear energy, the types of power reactor (thermal and fast reactors) commissioned or under development, the factors involved in processing the irradiated fuel elements, the part played by fast breeder reactors in conserving uranium supplies, and the storage of radioactive waste.

- 3851 (EURFNR--1418, pp 4-11) **THOUGHTS ON THE DEVELOPMENT OF FUEL ELEMENTS.** Karsten, G. Jan 1978. Translation by T.J. Clark of KFK--2416, Oct 1976.

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Fuel element of the sodium breeder.

Information concerning SNR fuel element development is presented. Fabrication processes and material questions are discussed.

- 3852 (EURFNR--1418, pp 12-18) **SNR-CORES: KNK II, SNR-300 TO SNR-2.** Hoechel, J. Jan 1978. Translation by T.J. Clark of KFK--2416, Oct 1976.

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Fuel element of the sodium breeder.

Design characteristics for the various SNR-1 reactor core configurations are presented.

- 3853 (EURFNR--1418, pp 19-26) **FUEL ELEMENTS FOR BREEDER POWER PLANTS: INTERNATIONAL STATUS.** Kummerer, K. Jan 1978. (Kernforschungszentrum, Karlsruhe, Ger.). Translation by T.J. Clark of KFK--2416, Oct 1976.

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Fuel element of the sodium breeder. The survey includes all fast breeder reactors which are already in operation, under construction, or in the concrete planning stage. Here a distinction can be made between research and test reactors (FT), prototype power reactors (PR) and actual power plant reactors (KR). The list gives more details (name, category, power, location, beginning of construction, full load) for the total of 21 fast breeders.

- 3854 (EURFNR--1418, pp 38-41) **CORE AND FUEL ELEMENT DESIGN FOR THE SNR-2.** Spenke, H.; Wehmann, U. (Interatom, Bensberg, Ger.). Jan 1978. Translation by T.J. Clark of KFK--2416, Oct 1976.

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Fuel element of the sodium breeder.

Specifications for the SNR-2 reactor fuel rods and core configuration are presented.

- 3855 (EURFNR--1418, pp 42-49) **CORE ELEMENTS DURING POWER VARIATIONS AND DISTURBANCES.** Toebe, H.; Gestermann, F. Jan 1978. Translation by T.J. Clark of KFK--2416, Oct 1976.

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Fuel element of the sodium breeder.

Design requirements for SNR-1 reactor fuel elements are described for normal and transient operating conditions.

3856 (EURFNR--1418, pp 61-71) STATUS OF CARBIDE FUEL ELEMENT DEVELOPMENT. Muehling, G. Jan 1978. Translation by T.J. Clark of KFK--2416, Oct 1976.

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Fuel element of the sodium breeder. The development of PuC--UC fuel for SNR-1 reactor is described. Results of irradiation tests on PuC--UC fuel rods are included.

3857 (EURFNR--1418, pp 73-82) CHEMICAL AND MECHANICAL INTERACTION IN THE OXIDE FUEL ROD. Thuemmler, F.; Dienst, W. (Kernforschungszentrum, Karlsruhe, Ger.). Jan 1978. Translation by T.J. Clark of KFK--2416, Oct 1976.

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Fuel element of the sodium breeder. An important consideration for the design of fuel rods is the prevention of damage to the cladding, and especially of fissures in the cladding up to the desired burn-up. Cladding, tube stresses leading to the formation of cracks can originate from inside through mechanical stresses on the cladding, in which a chemical reaction between the fuel and the cladding tube also plays an important and damaging part. Work on the chemical and mechanical interaction between the fuel and cladding material in fuel elements for fast breeder reactors is described.

3858 (EURFNR--1418, pp 83-90) STATE OF DEVELOPMENT OF CLADDING AND STRUCTURAL MATERIAL FOR SNR CORE ELEMENTS. Dietz, W. Jan 1978. Translation by T.J. Clark of KFK--2416, Oct 1976.

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Fuel element of the sodium breeder. The questions of high-temperature plasticity and the swelling which play an important part in the mechanical design of the core are studied. These questions must be treated generally for components and their materials in the region of the SNR core: Here there is a limitation to the materials and their metallurgical status, as they are specified for the cladding tube and fuel element shroud for the SNR-300 Mk-Ia. The experimental data come from the radiation tests illustrated. Temperature and neutron dosages are given as significant amounts.

3859 (EURFNR--1418, pp 91-97) FURTHER DEVELOPMENT OF CLADDING AND CORE STRUCTURAL MATERIALS. Anderko, K. Jan 1978. Translation by T.J. Clark of KFK--2416, Oct 1976.

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Fuel element of the sodium breeder. Tensile properties of Type 304 and Type 316 stainless steels are compared to 1.4970 austenitic steel. Physical radiation effects on fracture strength are discussed.

3860 (EURFNR--1418, pp 115-128) RE-TREATMENT OF FAST BREEDER FUEL ELEMENTS. Ochsenfeld, W.; Baumgaertner, F.; Schmieder, H. (Kernforschungszentrum, Karlsruhe, Ger.). Jan 1978. Translation by T.J. Clark of KFK--2416, Oct 1976.

From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R.

Germany (28 Oct 1976).

Fuel element of the sodium breeder. Reprocessing equipment and procedures for LMFR type reactor $\text{PuO}_2\text{--UO}_2$ fuel elements are described.

3861 PERFORMANCE TEST PROGRAM OF DASHPOT FOR FFTF COLD LEG CHECK VALVE. Sharma, D.K.; Henschel, R.; Krawchuk, M. (Foster Wheeler Corp., Livingston, N.J. (USA). John Blizard Research Center). Int. J. Pressure Vessels Piping; 6: No. 1, 41-52 (Jan 1978).

The performance tests that were carried out on the prototypical dashpot for the Fast Flux Test Facility Cold Leg Check Valve (FFTF-CLCV) are described and evaluated. The study consisted of acceptance testing in freon, argon and air, and final testing in liquid sodium. It was concluded that the dashpot performance was acceptable. This dashpot was previously developed through a test program that was primarily conducted in water.

3862 DASHPOT DEVELOPMENT TEST FOR FFTF COLD LEG CHECK VALVE. Sharma, D.K.; Henschel, R.; Krawchuk, M. (Foster Wheeler Corp., Livingston, N.J. (USA). John Blizard Research Center). Int. J. Pressure Vessels Piping; 6: No. 1, 23-39 (Jan 1978).

Work that was performed to develop the dashpot for the Fast Flux Test Facility Cold Leg Check Valve is presented. It comprised the water testing of two basic dashpot designs and their modifications. The two basic designs included an annular and a non-annular concept. The programme consisted of subjecting each dashpot to a series of tests in water to investigate its filling, recocking and damping behaviour. The test results discussed showed the non-annular design to be superior and to meet performance requirements in virtually all cases.

3863 EFFECTS OF STRESS ON SWELLING IN 316 STAINLESS STEEL. Bates, J.F.; Gilbert, E.R. (Hanford Engineering Development Lab., Richland, Wash. (USA)). J. Nucl. Mater.; 71: No. 2, 286-291 (Jan 1978).

Stress was found to increase the magnitude of irradiation-induced swelling in 316 stainless steel. Measurement of the densities of pressurized tube specimens, irradiated at temperatures of approximately 430-475°C to peak fluences of approximately 9×10^{22} n/cm² ($E > 0.1$ MeV) in EBR-II, has indicated increased swelling in both the annealed and 20% cold worked conditions of this alloy. Swelling in the annealed specimens was observed to increase linearly with hoop stress up to approximately 20 ksi (130 MPa), whereupon further increases in stress resulted in reduced swelling. Swelling in the cold worked material was linear with stress up to levels of approximately 28 ksi (193 MPa).

3864 INCREASED SWELLING IN 20% COLD WORKED 316 STAINLESS STEEL DUE TO INREACTOR TEMPERATURE REDUCTIONS. Bates, J.F.; Gelles, D.S. (Hanford Engineering Development Lab., Richland, Wash. (USA)). J. Nucl. Mater.; 71: No. 2, 365-367 (Jan 1978).

This letter describes the swelling behavior of test specimens of 20% cold worked 316 stainless steel when a reduction in temperature takes place during the course of irradiation in a fast reactor.

3865 GROWTH OF A SODIUM VAPOUR BUBBLE RISING IN THE SUPERHEATED LIQUID. Shah, V.L.

(Wisconsin Univ., Milwaukee (USA)); Sha, W.T. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 45: No. 1, 81-91 (Jan 1978).

The growth of a single spherical bubble moving in a uniformly superheated liquid is considered. The equations of motion and energy are presented in the forms which take into consideration the interaction between the motion and the growth. The fourth order Runge-Kutta method is used to obtain a simultaneous solution of equations of motion and growth rate. The solution is compared with the available experimental results. The results for liquid sodium are presented for a range of pressures and Jakob numbers.

3866 (EURFNR--1418) FUEL ELEMENT OF THE SODIUM BREEDER. Karsten, G. (ed.). (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Bruefer). Jan 1978. Translation by T.J. Clark of KFK--2416, Oct 1976 (CONF-7610135--). 132p. Dep. NTIS (US Sales Only), PC A07/MF A01. From KTG technical day on fuel element of the sodium-cooled breeder; Karlsruhe, F.R. Germany (28 Oct 1976).

Work performed under United States--Euratom Fast Reactor Exchange Program.

Separate abstracts are presented for 10 papers on fuel elements and core materials for LMFBR type reactors. Five papers have been previously abstracted and added to the data base.

3867 (EURFNR--1421) DESIGN, IRRADIATION, AND POST-IRRADIATION EXAMINATION OF NA-BONDED (U,Pu)C FUEL RODS OF THE TEST GROUP MOL 15. Steiner, H.; Freund, D.; Jacobi, O.; Weimar, P. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). Jan 1978. Translation by T.J. Clark of KFK--2451 169p. Dep. NTIS (US Sales Only), PC A08/MF A01.

Work performed under United States--Euratom Fast Reactor Exchange Program.

The Mol 15 experiment entails the irradiation of four fuel rod samples with UC/PuC fuel with various M_{23} proportions under Na bonding. In order to be able to make judgments among them on the development potential of fuel variants, it was necessary to get comparable irradiation results, which could be optimally realized in so-called "integrated" experiments: In an irradiation capsule, 4 short rods were tested under practically identical irradiation conditions. In order to obtain high burn-ups in a comparatively short time span, fuel rods with a small fuel rod diameter of 5.6 mm were used. A detailed description of this experiment is given in the present report. In addition to a representation of the design and the specification, the manufacture of the test samples, the course of the irradiation, as well as the non-destructive and destructive post-investigation are documented. Finally, the fuel swelling, the fuel structure which arises, and the Pu-redistribution are discussed in comparison with previous experiments.

3868 ABSORBER BALLS TESTED FOR LMFBR SECONDARY SHUTDOWN. Marshall, P. Nucl. Eng. Int.; 23: No. 266, 26-27 (Jan 1978).

The hydraulic performance of a column of tantalum alloy absorber balls and a flow shutoff valve which could be used as a fast breeder shutdown assembly is being evaluated in the Hydraulic Testing facility at the Liquid Metal Engineering Centre in California. The balls are designed to fall into the reactor core by gravity when the coolant flow is interrupted. A second concept involves and

articulated absorber with a high degree of chain-like flexibility. Both systems are intended for an Inherent Secondary Shutdown System for advance LMFBRs. Assemblies and tests are briefly described.

3869 THEORETICAL ANALYSIS APPLIED TO ELECTROMAGNETIC INSTRUMENTS USED IN LIQUID SODIUM. Farrington, B.J.; Hughes, G. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Nucl. Energy; 17: No. 1, 57-64 (Jan 1978).

The solution of Maxwell's equations to obtain the electric field distribution in a conductive, isotropic, inhomogeneous medium is performed using a finite difference approach. The initial application is devoted to the case of coils, carrying a low frequency sinusoidal driving current, which are surrounded by media having any desired combination of electrical conductivity and linear magnetic permeability. An estimate is made of the errors involved, particularly when using this technique to analyse electromagnetic sensors for use in liquid sodium. The results obtained when the developed program is used to solve a simple problem are compared with the corresponding analytical Bessel solution, proving the validity of the method. Expressions are derived to enable the calculation of flux density, induced voltage, coil impedance and the effect of flowing media from the computed values of electric field. Detailed analyses are made of an electromagnetic temperature sensor, constructed of stainless-steel-sheathed mineral-insulated cable, and of a flux distortion flow meter used in the Dounreay prototype fast reactor. Satisfactory agreement with the experimental performance of these sensors is obtained.

3870 DASHPOT DEVELOPMENT TEST FOR FFTF COLD LEG CHECK VALVE. Sharma, D.K.; Henschel, R.; Krawchuk, M. (Foster Wheeler Energy Corp, John Blizard Res Cent, Livingston, NJ). Int. J. Pressure Vessels Piping; 6: No. 1, 23-29 (Jan 1978).

This paper presents work that was performed to develop the dashpot for the Fast Flux Test Facility Cold Leg Check Valve. The portion covered herein involved the water testing of two basic dashpot designs and their modifications. The program consisted of subjecting each dashpot to a series of tests in water to investigate its filling, recocking and damping behavior. The test results discussed herein showed the nonannular design to be superior and to meet performance requirements in virtually all cases. 6 refs.

3871 PERFORMANCE TEST PROGRAM OF DASHPOT FOR FFTF COLD LEG CHECK VALVE. Sharma, D.K.; Henschel, R.; Krawchuk, M. (Foster Wheeler Energy Corp, Livingston, NJ). Int. J. Pressure Vessels Piping; 6: No. 1, 41-52 (Jan 1978).

This paper describes and evaluates the performance tests that were carried out on the prototypical dashpot for the Fast Flux Test Facility Cold Leg Check Valve (FFTF-CLCV). The study consisted of acceptance testing in freon, argon and air, and final testing in liquid sodium. Based upon this work it was concluded that the dashpot performance was acceptable. This dashpot was previously developed through a test program that was primarily conducted in water. 5 refs.

3872 ROLE OF FISSION PRODUCTS IN WHOLE-CORE ACCIDENTS IN FAST REACTORS. Baker, A.R.

(UKAEA Fast Reactor Development Directorate, Risley); Teague, H.J. (UKAEA Safety and Reliability Directorate, Culcheth). Nucl. Energy; 17: No. 1, 31-40 (Jan 1978).

Revised version of paper presented at IAEA Specialists' Meeting at Harwell, June 1977. Also available as ND-R-4(R).

The review of the role of fission products in whole-core accidents falls into two parts. Firstly, there is a discussion of the hypothetical accidents usually considered in the UK, and of how they are dealt with. Secondly, individual topics, where fission products are known to be important or might be so, are discussed. There is a brief discussion of the UK work on the establishment of an equation of state for unirradiated fuel and on how this might be extended to incorporate fission product effects. An important issue is the contribution of fission products to the effective vapour pressure and the experimental programme on the pulsed reactor VIPER investigates this. Fission products may influence the probability of occurrence and the severity of interactions between molten fuel and coolant. Finally, the fission product effects in the pre-disassembly, disassembly, and recriticality stages of an accident are discussed.

3873 TEMPERATURE FLUCTUATION OF SODIUM IN ANNULAR FLOW CHANNEL HEATED BY SINGLE-PIN WITH BLOCKAGE. Miyazaki, K.; Kimura, J.; Ogawa, M.; Okada, T. (Osaka Univ, Jpn). J. Nucl. Sci. Technol. (Tokyo); 15: No. 1, 10-23 (Jan 1978).

Root mean square (RMS) value and power spectral density (PSD) of temperature fluctuation were measured with use of forced-circulating sodium in an annular channel (6.5 mm I.D., 20 mm O.D.) with a concentric disk to simulate blockage (about 80%) of sodium flow. The RMS value is proportional to the wall-to-bulk-fluid temperature difference in heat transfer, presenting the similar dependence on the heat flux and flow velocity. The fluctuations of temperature are greatly attenuated in the upper unheated section where the radial temperature gradient is absent, and consequently it is suggested that the fluctuations of temperature should be caused by the local turbulence of flow, such as a vortex street due to blockage in the present experiment, under the presence of large gradient of temperature near the heating surface. 13 refs.

3874 AIM, CONSTRUCTION AND OPERATION OF THE FAST-BREEDER NUCLEAR POWER STATION IN KALKAR (SNR-300). Schoenig, R. Braunkohle; 30: No. 1, 11-19 (Jan 1978). (In German).

From Lecture during the 43rd meeting of the DEBRIV committee Feuerungstechnik und Waermewirtschaft; Koeln, Germany, F.R. (21 Oct 1977).

After many years of research and development work the governments of the Federal Republic of Germany, Belgium and Holland decided in 1972 to promote the construction of a prototype power plant with a sodium-cooled fast-breeder reactor with an electrical capacity of approx. 300 MW. Experience is to be gained with this plant in order to determine in how far this type of reactor can be used for the commercial generation of power. The FBR can contribute on the one hand to closing the energy gap becoming apparent in the future and on the other hand to ensuring some 60 times better utilisation of existing U reserves. Furthermore, it also offers an important contribution to solving the depollution problem as it not only converts the U^{235} - which is a non-fissionable matter in LWR and which occurs to about 99% in natural U -

into fissile Pu, but also uses the Pu generated in the reactors, removing it from the ecological cycle. All tests have shown that the FBR has been mastered according to present-day engineering standards and that the operation of FBR brings no increased dangers for population or environment compared with other types of nuclear power stations. The construction of the prototype power plant in Kalkar has begun in 1973. It is expected that it will be completed and commissioned by 1983.

3875 TEMPERATURE FLUCTUATION OF SODIUM IN ANNULAR FLOW CHANNEL HEATED BY SINGLE-PIN WITH BLOCKAGE. Miyazaki, K.; Kimura, J.; Ogawa, M.; Okada, T. (Osaka Univ., Suita (Japan). Faculty of Engineering). J. Nucl. Sci. Technol. (Tokyo); 15: No. 1, 10-23 (Jan 1978).

Root mean square (RMS) value and power spectral density (PSD) of temperature fluctuation were measured with use of forced-circulating sodium in an annular channel (6.5 mm I.D., 20 mm O.D.) with concentric disk to simulate blockage (about 80%) of sodium flow. The experimental range of the heat flux was 40 - 150 W/cm² and the bulk flow velocity 0.14 - 0.41 m/sec ($Re = 7.7 \times 10^3 - 2.3 \times 10^4$) under a temperature of 500 - 800°C. The RMS value measured at the exit of heating section (150 mm downstream from the blockage) is larger by a factor of 2 - 3 than that in the wake (10 - 20 mm downstream from the blockage), marking a few deg.C for a heat flux of 105 W/cm² and a flow velocity of 0.27 m/sec. The RMS value is proportional to the wall-to-bulk-fluid temperature difference in heat transfer, presenting the similar dependence on the heat flux and flow velocity. The fluctuations of temperature are greatly attenuated in the upper unheated section where the radial temperature gradient is absent, and consequently it is suggested that the fluctuations of temperature should be caused by the local turbulence of flow, such as a vortex street due to blockage in the present experiment, under the presence of large gradient of temperature near the heating surface.

3876 CRACK GROWTH BEHAVIOUR OF THE AUSTENITIC STEELS X 6 CRNI 18 11 AND X 6 CRNIMO 17 13 UNDER THE CONDITIONS OF A NATRIUM-COOLED FAST BREEDER REACTOR. Stahlberg, R. (Technischer Ueberwachungs-Verein Rheinland e.V., Koeln (Germany, F.R.). Inst. fuer Materialpruefung und Chemie). Arch. Eisenhuettenwes.; 49: No. 1, 1-8 (Jan 1978). (In German).

Mathematical description of the crack growth behaviour of the steels X 6 CrNi 18 11 containing 0.06% C, 18% Cr and 11% Ni and also X 6 CrNiMo 17 13 containing 0.06% C, 17% Cr, 2.5% Mo and 13% Ni in air at temperatures up to 646°C and stress cycle frequencies of $8.3 \cdot 10^{-2}$ to $4.1 \cdot 10^3$ cycles/min. Influence of carbide precipitates. Comparison of the effect of thermal and fast neutrons. Frequency dependence of the crack propagation rate. Delayed crack propagation in the neighbourhood of welded joints. Crack growth under vacuum and in liquid sodium between 427 and 538°C.

3877 CERTAIN ASPECTS OF ALTERNATIVE CORE DISRUPTION ACCIDENTS IN LMFBRS. Chan, C.K.; Min, T.K.; Okrent, D. (California Univ., Los Angeles (USA). Dept. of Chemical, Nuclear, and Thermal Engineering). Nucl. Eng. Des.; 51: No. 2, 263-272 (Jan 1978).

This paper examines the initial course of a postulated accident scenario in an LMFB, involving the rupture of all piping connected to the reactor vessel, in the event of an earthquake (or an equivalent scenario involving

both loss of heat removal and system rupture). The core is successfully shut down, but decay heat imposes a threat to core integrity. At the onset of pipe rupture, the fuel elements are cooled by natural circulation, followed by subcooling and nucleate boiling. Continuation of sodium evaporation leads to core dryout, clad melting, and subassembly wall failure. Clad melting is found to occur at a location close to the top of the core at a rate of 0.08 ft³/s. The sodium vapor velocity is not high enough to carry the molten steel to the upper blanket region; therefore, flow-channel blockage in the lower axial blanket region is expected. At the time of clad melting, the fuel temperature rises by approximately 20°F/s, while the temperature-rise rate at the can wall, due to heat radiation from the fuel pin, is 10°F/s. Failure of the can wall initiates gross fuel motion. Continuation of the heat generation in the fuel pellet leads to the melting of the control rod support. Both the fuel motion and the control rod failure mark the start of reactivity insertion.

- 3878 RESEARCH AND DEVELOPMENT ACTIVITIES RELATIVES TO THE REACTOR VESSEL. Benoist, E. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides). Bull. Inf. Sci. Tech. (Paris); No. 227, 99-110 (Jan-Feb 1978). (In French).

The Super Phenix vessel is very similar in design to that of Phenix. In this way it has been possible to benefit from past experience and to reduce the number of preliminary tests needed for the definition of equipment. However the increase diameter of the reactor vessel has led to the evolvement, for certain components, of new solutions requiring development trials. Conversely other components technologically satisfactory will only undergo qualification tests.

- 3879 STUDIES AND FABRICATION OF THE FUEL. Chaumont, J.M.; Wallet, P. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides); Grignon, F. (Compagnie Generale des Matieres Nucleaires (COGEMA), 92 - Chatillon (France)). Bull. Inf. Sci. Tech. (Paris); No. 227, 63-80 (Jan-Feb 1978). (In French).

In this article the general characteristics of the fuel subassemblies in the SUPER PHENIX reactor are presented, and a detailed description is given of the various types of core subassemblies: fissile, fertile, control and shielding subassemblies. The principles guiding the studies and design of the different types of subassemblies are exposed, and the technological limitations adopted for fuel element operation are given. The various types of damage leading to these limitations are reviewed, and reference is made to operation experience gained from prototype reactors RAPSDIE and PHENIX. The organization of the fuel element fabrication for SUPER PHENIX is then presented. The various steps in this fabrication are reviewed, and the quality control organization is described. The capacity of the plants and the foreseen agenda for fabrication of fuel subassemblies are given. The fabrication of fertile subassemblies and other core components is then shortly described.

- 3880 DESIGN AND CONSTRUCTION OF THE NUCLEAR BOILER. Bull. Inf. Sci. Tech. (Paris); No. 227, 39-61 (Jan-Feb 1978). (In French).
The Creys-Malville station is equipped with

a nuclear boiler of the Super Phenix 1 type, delivering a thermal power of 3000MW and possessing a sodium-cooled fast neutrons reactor. The options taken continue in the direct line of the Phenix boiler technology, which means that it was possible to benefit largely from experience of the building start-up and piloting of this station. All equipment and components of the nuclear boiler except the steam generators and their associated auxiliary circuits are housed in the reactor and handling building. The steam generators and auxiliaries are built on in a star formation to the outside of the reactor building enclosure. The sodium-cooled fast breeder reactor is of the 'integrated primary circuit' type. This means that the whole primary sodium circuit passing through the core is placed in the main reactor vessel with its four main pumps and eight intermediate exchanges which transfer calories from the core to four outer and independent secondary loops. The reactor is planned to start up during the year 1983.

- 3881 RESEARCH AND DEVELOPMENT ACTIVITIES CONCERNING THE SUPER-PHENIX CORE. Ravier, J.; Sicard, B.; Estavoyer, M. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides). Bull. Inf. Sci. Tech. (Paris); No. 227, 111-117 (Jan-Feb 1978). (In French).

A research and development programme supplies the necessary guarantees to solutions adopted for the Super Phenix core design. Studies are in progress to examine the start-up pattern, test programme management and follow-up of the fuel assemblies.

- 3882 CREYS-MALVILLE POWER STATION. Robert, E.; Rodet, E.; Lucenet, G. (Societe Centrale Nucleaire Europeenne a Neutrons Rapides (NERSA), 69 - Lyon (France)). Bull. Inf. Sci. Tech. (Paris); No. 227, 11-32 (Jan-Feb 1978). (In French).

This article reviews the main characteristics and essential options held for this station. The following points are examined in turn: the context chosen for its construction (especially at the level of the European Electricity Producer's Association); essential data concerning the Creys Malville site (general situation, climatology, hydrology, geology and seismology ...); major options relative to the running of the station; specifications of the machine room and the power evacuation water-steam circuit; system of confinement by consecutive barriers; certain safety options (disturbances from outside or inside) directly affecting the design of the buildings; origin and nature of gaseous effluents and liquid wastes from a plant of this kind. The state of advancement of the project is also summed up: administration procedures, markets and building work.

- 3883 STUDIES AND RESEARCH RELATIVES TO THE SAFETY OF THE SUPER-PHENIX PROJECT. Anselin, F.; Penet, F. Bull. Inf. Sci. Tech. (Paris); No. 227, 119-139 (Jan-Feb 1978). (In French).

The analysis of safety reports concerning the Creys Malville power station (Superphenix) must be based on technical data supplied by the NERSA, responsible for the plant, and on results of research and development programme carried out in various establishments and at the CEA in particular. By virtue of the procedure laid down for the safety analysis, i.e. analysis by the barrier method, verification of reactor shut down rules at power, permanence of cooling, confinement of

dangerous products in the event of hypothetical failure of the above two functions safety R and D programmes have a double aspect: accident prevention on the one hand and study of the development of accident, even the most hypothetical on the other. In the accident prevention field the studies deal with the resistance of barriers under normal and accidental working conditions, inspection systems and reactor safety functions allowing abnormal situations to be detected and the reactor shut down; whence the special emphasis placed on emergency shut-down and cooling systems. In the accident field the R and D activities cover a wide range of studies on phenomena liable to arise, independently of their probability of occurrence during the lifetime of the reactor; heating in the mass or boiling of sodium, fuel, meeting, movements of fused materials, fuel-sodium thermal interaction, core deformation, resistance of confinement recovery of molten fuel, post-accident cooling, transfer of radioactivity and contamination outside the reactor, radiological consequences and means of confinement of dangerous products.

- 3884 MANIPULATION OF SCIENTIFIC DATA FOR NUCLEAR ENERGY CALCULATIONS. Montgomery, K.R. pp 160-165 of Generalized data management systems and scientific information. Paris; OECD Nuclear Energy Agency (1978).

From OECD/NEA working group on nuclear information; Berkeley, CA, USA (5 Oct 1977).
See CONF-771062--.

Fuel management strategy is vital to the efficient operation of the PFR at Dounreay. The strategy encompasses planning, operational, and post-operational activities. Fuel management calculations are outlined for these stages of operation. The unusual nature of the associated problems, the ideal requirements of the system to be adopted, and the present technique for their solution are given. Some of the essential requirements of a scientific data base to support these calculations are defined. (RWR)

- 3885 DATABANK FOR THE PROTOTYPE FAST REACTOR. Montgomery, K.R. pp 250-256 of Generalized data management systems and scientific information. Paris; OECD Nuclear Energy Agency (1978).

From OECD/NEA working group on nuclear information; Berkeley, CA, USA (5 Oct 1977).
See CONF-771062--.

This databank is a large set of files controlled by a management suite of CQBOL data handling programs. The data, data dictionary and data directory are stored on exchangeable disc packs in the Random Access Index Sequential mode on an ICL 4/72 computer at the UKAEA headquarters at Risley. It is a databank for the UKAEA Prototype Fast Reactor which stores data pertaining to the entire composition of the in-core components, concentrating particularly upon the fissile fuelled driver assemblies and the many experimental assemblies of both fuel pin and cladding/structural materials. When the reactor is operating at full power the databank will contain reactor operating history and reactor physics data, and also post-irradiation examination data for the discharged units. The databank will represent a complete data compilation of in-reactor performance of fully documented materials. The current size of the databank is of the order of 100 megabytes and is expected to increase to approximately 200 megabytes over the next 2 years.

SWELLING OF THE STEEL 1.4970 IN THE RAPSDIE CLUSTER EXPERIMENT. Ehrlich, K.; Schneider, W. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). pp 617-619 of Reactor congress, Hannover, 4.4.-7.4.1978. Section 4: fuel elements and fuel element materials. Eggenstein-Leopoldshafen, Germany, F.R.; ZAED (1978). (In German)

From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

DENSITY;FAST NEUTRONS;HIGH TEMPERATURE; PHYSICAL RADIATION EFFECTS;POROSITY;LMFBR TYPE REACTORS;REACTOR MATERIALS;CHROMIUM-NICKEL STEELS;SWELLING

- 3887 CAN SWELLING IN FAST-BREEDER FUEL RODS WITH OXIDIC AND CARBIDIC FUEL. Mueller-Lyda, I. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). pp 668-671 of Reaktortagung, Hannover, 4.4.-7.4.1978. Sektion 4. Brennelemente und Brennelement-Werkstoffe. Eggenstein-Leopoldshafen, Germany, F.R.; ZAED (1978). (In German)

From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

CREEP;DEFORMATION;LMFBR TYPE REACTORS;FUEL CANS;FUEL ELEMENTS;FUEL-CLADDING INTERACTIONS; NEUTRONS;PHYSICAL RADIATION EFFECTS;PLUTONIUM CARBIDES;PLUTONIUM OXIDES;URANIUM CARBIDES; URANIUM OXIDES

- 3888 EFFECT OF ENGINEERED SAFETY FEATURES ON THE RISK OF HYPOTHETICAL LMFBR ACCIDENTS. Cybulskis, P. (Battelle Columbus Labs., OH). Nucl. Saf.; 19: No. 2, 190-204(1978).

The risks of hypothetical core-disruptive accidents in liquid-metal-cooled fast breeder reactors which involve meltthrough of the reactor vessel are compared for two plant designs: one design without specific provisions to accommodate such an accident and the other design with an ex-vessel core catcher and a cavity hot liner. The approach to risk analysis used is that developed in the Reactor Safety Study (WASH-1400). Since the probability of occurrence of such an event has not been evaluated, however, insight into the potential risk is gained only on a relative basis. The principal conclusions of this study are: (1) adding a core catcher--hot liner reduces the probability of accidents having major consequences; (2) the degree to which hot liner--core catcher systems can reduce the risk of melt-through accidents is limited by the failure probability of these systems; (3) fractional radioactive releases to the environment in the liquid-metal-cooled fast breeder reactor accidents considered are comparable to those from the light-water reactors evaluated in WASH-1400; (4) since sodium--concrete reactions are a dominant driving force during the accident, the integrity of the cavity liner is as important as the function of the core catcher; (5) there may be other accidents or paths to radioactive releases that are not affected by the addition of a hot liner--core catcher.

- 3889 CLINCH RIVER BREEDER REACTOR PLANT SAFETY STUDY. Piper, H.B.; Conradi, L.L.; Buhl, A.R.; Wood, P.J.; Leaver, D.E.W. Nucl. Saf.; 19: No. 3, 316-329(1978).

The article presents a review and discussion of the objectives, methods, techniques, and results of a safety study that was conducted for the Clinch River Breeder Reactor (CRBR) Plant. The objectives of the study were: (1) to provide a realistic assessment of accident risks to the public associated with operation

of the CRBR, (2) to place these identified risks in perspective with other local societal risks, and (3) to aid in determining whether accident risks from the CRBR are comparable to those of previously licensed reactors. Achievement of the objectives of the study has required identification of significant contributors to risk in a logical and orderly manner. Consideration of a comprehensive set of accident initiators, inclusion of experience data, reliance on proven methods and techniques, evaluation of a wide range of radionuclide releases and associated health effects, and utilization of experienced risk analysis are the salient elements employed in the systematic approach to this study. This, together with heavy reliance on experience gained during years of LWR design, licensing, and operation, provides reasonable assurance that the study objectives have been achieved. The results of the CRBR Plant Safety Study indicate that the risk arising from the operation of the CRBR Plant is small in comparison to other local societal risks and that the risk from the CRBR Plant is comparable to the risk from previously licensed nuclear power plants, as identified in the Reactor Safety Study.

- 3890 SODIUM LOOP SAFETY FACILITY. Gartside, C.H.; Bezella, W.A.; Thompson, D.H.; Lennox, D.H.; Tessier, J.H. (Argonne National Lab., IL). Nucl. Saf.; 19: No. 3, 339-354(1978).

As the only facility in the United States capable of conducting safety tests on relatively large, full-length liquid-metal fast breeder reactor (LMFBR) fuel assemblies at steady-state power levels, the Sodium Loop Safety Facility (SLSF) is an important part of the LMFBR Safety Program. The in-pile experiments that began in September 1975 in the Engineering Test Reactor (ETR) are providing data needed for assessment of critical safety questions. Presented in the article are descriptions of the facility, its experimental capabilities, and the overall research program.

- 3891 CREEP BEHAVIOUR OF SNR CLADDING TUBE MATERIAL UNDER THE TEMPERATURE TRANSIENTS RELEVANT TO COOLING DISTURBANCES. Preininger, D.; Anderko, K. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). pp 613-616 of Reaktortagung, Hannover, 4.4.-7.4.1978. Sektion 4: Brennelemente und Brennelement-Werkstoffe. Eggenstein-Leopoldshafen, Germany, F.R.; ZAED (1978). (In German)
From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).
CREEP; FUEL CANS; LOSS OF COOLANT; SIMULATION; SNR-1 REACTOR; TEMPERATURE NOISE; THERMAL CYCLING; TRANSIENTS; REACTOR MATERIALS; CHROMIUM-NICKEL STEELS; TITANIUM ADDITIONS

- 3892 INVESTIGATION OF THE DYNAMIC BEHAVIOUR OF DETECTORS FOR GAS BUBBLES IN SODIUM FLOWS. Benkert, J.; Raes, K.H.; Reimche, W.; Stegemann, D. (Technische Univ. Hannover (Germany, F.R.). Inst. fuer Kerntechnik). pp 178-181 of Reaktortagung, Hannover, 4.4.-7.4.1978. Sektion 1: Reaktorauslegung. Eggenstein-Leopoldshafen, Germany, F.R.; ZAED (1978). (In German)
From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).
BOILING; BUBBLES; LIQUID FLOW; MEASURING INSTRUMENTS; SENSITIVITY; SODIUM; LMFBR TYPE REACTORS; IN CORE INSTRUMENTS; PRIMARY COOLANT CIRCUITS; BOILING DETECTION

- 3893 CORE DESIGN OF SODIUM-COOLED FAST BREEDERS (SNR) WITH REGARD TO STRATEGIC DESIGN TARGETS. Stanculescu, A.; Wehmann, U.; Lukas, H.; Spenke, H.; Heinecke, J. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp 1051-1054 of Reaktortagung, Hannover, 4.4.-7.4.1978. Sektion 8: Reaktor- und Energiesysteme, Wirtschaftlichkeitsfragen. Eggenstein-Leopoldshafen, Germany, F.R.; ZAED (1978). (In German)

From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

On the basis of the discussions on the different possibilities of nuclear power generation which have lately been carried out with more fervour, the effects of some SNR core design parameters on the major strategic orientation parameters are studied in this paper. This is only a part of extensive investigations with the aim to determine the optimum core design parameters for the strategy assumed (for technical, economic, or political reasons).

- 3894 CHEMICAL BEHAVIOUR OF MIXED-CARBIDE FUEL RODS UNDER IRRADIATION. Kleykamp, H. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). pp 660-663 of Reaktortagung, Hannover, 4.4.-7.4.1978. Sektion 4: Brennelemente und Brennelement-Werkstoffe. Eggenstein-Leopoldshafen, Germany, F.R.; ZAED (1978). (In German)
From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).
LMFBR; BONDING; BR-2 REACTOR; CARBON; CARBURIZATION; CHEMICAL COMPOSITION; CHEMISTRY; DFR REACTOR; DIFFUSION; FISSION PRODUCTS; FUEL CANS; FUEL PINS; HELIUM; IRRADIATION; KRYPTON; MIXTURES; OXYGEN; PLUTONIUM CARBIDES; POROSITY; CHEMICAL RADIATION EFFECTS; SEPARATION PROCESSES; SODIUM; SPATIAL DISTRIBUTION; SWELLING; URANIUM CARBIDES; XENON; LMFBR TYPE REACTORS

- 3895 EFFECTS OF FISSION GAS BUBBLES IN THE BOND ON THE TEMPERATURE DISTRIBUTION IN NA-BONDED CARBIDE FUEL RODS. Steiner, H. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). pp 664-667 of Reaktortagung, Hannover, 4.4.-7.4.1978. Sektion 4: Brennelemente und Brennelement-Werkstoffe. Eggenstein-Leopoldshafen, Germany, F.R.; ZAED (1978). (In German)
From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).
BONDING; BUBBLES; CARBIDES; FISSION PRODUCTS; FUEL ELEMENTS; FUEL PELLETS; MATHEMATICAL MODELS; MICROSTRUCTURE; SODIUM; SPATIAL DISTRIBUTION; VERY HIGH TEMPERATURE; LMFBR TYPE REACTORS; TEMPERATURE DISTRIBUTION

- 3896 SPECIAL FEATURES OF THE THERMOHYDRAULIC DESIGN OF CONTROL/SCRAM AND SECONDARY SHUTDOWN ELEMENTS OF THE SNR 300. Coors, D. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp 118-121 of Reaktortagung, Hannover, 4.4.-7.4.1978. Sektion 1: Reaktorauslegung. Eggenstein-Leopoldshafen, Germany, F.R.; ZAED (1978). (In German)
From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).
- Element construction, flow rate distribution and thermohydraulic element layout;
- design data of the control elements for given input data.

- 3897 COMPUTER CODE SEURBNK-2 FOR FAST

REACTOR CONTAINMENT SAFETY STUDIES. Cameron, I.G.; Hankin, B.C.; Warham, A.G.P. (UKAEA Weapons Group, Aldermaston. Atomic Weapons Research Establishment); Benuzzi, A.; Yerkess, A. (Commission of the European Communities, Ispra (Italy). Joint Research Centre). Comput. Phys. Commun.; 13: No. 5, 297-310(1978).

The paper describes the theory and capabilities of the SEURBNUK-2 fluid flow code developed by JRC Ispra and AWRE Aldermaston for use in containment analysis for fast reactor safety studies. It is a 2D Eulerian code and is a major extension of the SURBOUM code developed earlier by Belgonucleaire and Aldermaston. SEURBNUK-2 has a full thin shell treatment for tanks of arbitrary shape and is able to include the effects of compressibility of the fluid. To check on the codes treatment of compressibility, results are presented for SEURBNUK calculations of an Ispra test shot 114 for the expansion of a Belgonucleaire charge in an overstrong water filled vessel. Comparisons are made both with experimental results and with calculations done on the 2D Lagrangian code REXCO-H and ASTARTE-2.

3898 TRANSFER FUNCTION OF TEMPERATURE FLUCTUATIONS FOR LIQUID METAL SLUG FLOW. Bunsch, J. (Eidgenössische Technische Hochschule, Zurich (Switzerland). Inst. fuer Reaktortechnik). Atomkernenergie; 31: No. 1, 23-27(1978).

The axial transport of temperature fluctuations in vertical slug flow is investigated. The behaviour of the temperature fluctuations in the high frequency range is well described only if heat conduction as a dissipation mechanism is taken into account. In the case of high thermal conductivity of the fluid a simple approximation for the transfer function of temperature fluctuations is found. Its form explains why the fluid velocity may be derived from the peak of temperature cross-correlation functions. Numerical results for the transfer function are given for liquid sodium. The influence of the axial conduction on the determination of heat transfer coefficient from temperature transfer functions is discussed.

3899 SODIUM-NAK ENGINEERING HANDBOOK. VOLUME III. SODIUM SYSTEMS, SAFETY, HANDLING, AND INSTRUMENTATION. Foust, O.J. (ed.). New York; Gordon and Breach, Science Publishers, Inc. (1978). 346p. (TID--28754).

The handbook is intended for use by present and future designers in the Liquid Metals Fast Breeder Reactor (LMFBR) Program and by the engineering and scientific community performing other type investigation and experimentation requiring high-temperature sodium and NaK technology. The arrangement of subject matter progresses from a technological discussion of sodium and sodium-potassium alloy (NaK) to discussions of various categories and uses of hardware in sodium and NaK systems. Emphasis is placed on sodium and NaK as heat-transport media. Sufficient detail is included for basic understanding of sodium and NaK technology and of technical aspects of sodium and NaK components and instrument systems. Information presented is considered adequate for use in feasibility studies and conceptual design, sizing components and systems, developing preliminary component and system descriptions, identifying technological limitations and problem areas, and defining basic constraints and parameters.

3900 FLOW AND MIXING CONDITIONS IN THE UPPER

PLENUM OF THE SNR 300. Benemann, A.; Granner, N.; Penny, W.; Schmidt, H.; Voj, P. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp 126-129 of Reaktortagung, Hannover. Sektion 1. Reaktorauslegung. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978). (In German)
From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

The sodium-cooled nuclear reactor is characterized by a relatively large upper tank plenum within which phenomena of energy and material transport determine the temperature and velocity distributions in the hot legs of the primary loop. During steady-state as well as transient loads this flow volume has the function of a mixing chamber for homogenizing the flow conditions and the sodium temperatures. Decisive for the mixing process are the different forces acting on the mass transport respectively the relations between them. During transient phenomena with greatly changing density conditions pressure gradients, forming on account of the differential densities, possibly may not be reduced because of the inertia forces of the flow, thus subvolumes of the upper plenum not taking part any more in the mixing process.

3901 CONCEPT OF THE PRIMARY HEAT TRANSFER SYSTEMS OF THE SNR 2. Berkenkamp, H.; Ratzel, W. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp 845-848 of Reaktortagung, Hannover. Sektion 5. Komponenten und Komponenten-Werkstoffe, Qualitaetssicherung. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978). (In German)
From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

PLANNING; PRIMARY COOLANT CIRCUITS; REACTOR VESSELS; SNR-2 REACTOR; TANKS

3902 THEORETICAL INTERPRETATION OF BOILING PHENOMENA IN ELECTRICALLY HEATED, SODIUM-COOLED 7-ROD ARRAYS. Bottoni, M.; Struwe, D. (Kernforschungszentrum Karlsruhe (Germany, F.R.)). Inst. fuer Reaktorentwicklung. pp 154-157 of Reaktortagung, Hannover. Sektion 1. Reaktorauslegung. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978). (In German)

From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

After describing the experimental boundary conditions their theoretical simulation is explained. Then theoretical and experimental results are compared.

3903 EXPERIMENTAL EVIDENCE OF THE EFFECT OF PERIPHERAL OBSTRUCTIONS ON THE TEMPERATURE FROM FACTOR OF A SPIRAL WIRE BUNDLE. Cornet, G.; Lamotte, G.; Verbiest, R. (Societe Belge pour l'Industrie Nucleaire, Brussels); Heinecke, J. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp 114-117 of Reaktortagung, Hannover. Sektion 1. Reaktorauslegung. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978).

From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

The above experiments show that - for geometry conditions (mainly pin diameter and pin pitch) typical of the SNR-300 Mk-II core - the introduction of filling wires at the

periphery of the bundle leads to improvements of the temperature form factor.

model of the SNR-300 fuel element cross-flowed with sodium. The investigations and findings so far are reported on.

- 3904 PRIMARY CIRCUIT THERMAL HYDRAULIC ANALYSIS OF SUPER PHENIX 1 REACTOR; SODIUM AND STRUCTURES TEMPERATURES CALCULATIONS; SALT WATER TESTS - HOT WATER TESTS. Kung, J.P.; Lambert, J. (Novatome Industries, 92 - Le Plessis Robinson (France)); Taillifet, D. (Electricite de France, 78 - Chatou. Lab. National d'Hydraulique); Ladet, J.; Gesl, E.; Marechal, A.; Vidil, R. (CEA Centre d'Etudes Nucleaires de Grenoble, 38 (France)). pp 122-125 of Reaktortagung, Hannover. Sektion 1. Reaktorauslegung. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978).

From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

In the case of SUPER-PHENIX, in view of both the difficulties involved in the apprehension of sodium temperatures under turbulent flow conditions, with forced, natural or mixed convection and the lack of accurate data on the characteristics of certain stainless steels at temperatures between 500°C and 600°C, it is clear that extensive means of investigation must be developed, including both computer programs and test facilities. Present results would tend to indicate that the choice of a double vessel system, using two vessels separated by a sodium filled cavity acting as a buffer between the hot and cold collectors, would provide a satisfactory solution from the technological viewpoint. With this type of set-up, the inner vessel acts essentially as a heat shield whereas the outer vessel bears the weight of the surrounding structures: It is required to withstand certain primary stresses but remains relatively cool.

- 3905 TESTING OF COMPONENTS OF THE HEAT TRANSFER SYSTEMS OF SODIUM-COOLED FAST BREEDERS IN A LARGE-SCALE TEST FACILITY. Mendte, K.; Jansing, W. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp 923-926 of Reaktortagung, Hannover. Sektion 6. Bau und Betrieb von Kernenergieanlagen. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978). (In German)

From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

HEAT EXCHANGERS; LIQUID METALS; MOCKUP; PERFORMANCE TESTING; PRIMARY COOLANT CIRCUITS; PUMPS; SNR-2 REACTOR; SODIUM; TEST FACILITIES; LMFBR TYPE REACTORS

- 3906 EXPERIMENTAL DETERMINATION OF LOCAL TEMPERATURE FIELD VARIATIONS DUE TO SPACER GRIDS IN THE CLADDING TUBES OF A ROD CLUSTER FLOWED THROUGH BY SODIUM. Moeller, R.; Tachoeke, H. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Reaktorbauelemente). pp 110-113 of Reaktortagung, Hannover. Sektion 1. Reaktorauslegung. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978). (In German)

From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

If spacer grids are used to keep the fuel rods in their places - as in the fuel elements of the SNR series, exact tests are necessary to find out whether and to what extent temperature peaks near the supporting points affect cladding tube design. To clarify this special problem, experimental investigations have been carried out for the first time in a rod cluster

- 3907 HANDLING OF SPENT FUEL ELEMENTS AT THE KALKAR NUCLEAR POWER STATION. Moryson, H.; Katzmann, H.; Pietz, H.U. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp 939-942 of Reaktortagung, Hannover. Sektion 6. Bau und Betrieb von Kernenergieanlagen. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978). (In German)

From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

MATERIALS HANDLING; PURIFICATION; REACTOR FUELING; REMOTE HANDLING; SNR-1 REACTOR; SPENT FUEL ELEMENTS; SPENT FUEL STORAGE; TRANSPORT

- 3908 ONE ASPECT OF THE CONSTRUCTION OF SUPERPHENIX; THE ON-SITE ASSEMBLY OF CERTAIN COMPONENTS OF THE NUCLEAR BOILER. Plessa, E.; Lallement, S. pp 996-997 of Reaktortagung, Hannover. Sektion 6. Bau und Betrieb von Kernenergieanlagen. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978).

From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

CONSTRUCTION; CRANES; FABRICATION; MATERIALS HANDLING; PERFORMANCE; PHENIX REACTOR; REACTOR COMPONENTS

- 3909 CREYS-MALVILLE POWER PLANT (SUPER PHENIX): THE MAIN OPTIONS OF THE REACTOR BUILDING VENTILATION SYSTEM. Plessa, A.; Rodet, E.; Garberini, C. (Societe Nucleaire Europeenne a Neutrons Rapides (NERSA), 69 - Lyon (France)). pp 841-844 of Reaktortagung, Hannover. Sektion 5. Komponenten und Komponenten-Werkstoffe, Qualitaetssicherung. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978).

From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

CONTAINMENT BUILDINGS; PLANNING; REACTOR VESSELS; SUPER PHENIX REACTOR; VENTILATION

- 3910 COURSES OF ACCIDENTS IN SODIUM-HEATED STEAM GENERATORS AND THEIR CONTROL, USING THE SNR-300 AS AN EXAMPLE. Dumm, K. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp 833-836 of Reaktortagung, Hannover. Sektion 5. Komponenten und Komponenten-Werkstoffe, Qualitaetssicherung. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978). (In German)

From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

FAILURES; H CODES; HYDROGEN; LEAK DETECTORS; LEAKS; MOLTEN METAL-WATER REACTIONS; SAFETY ENGINEERING; SNR-1 REACTOR; SODIUM; STEAM; STEAM GENERATORS

- 3911 SNR 2 - INTERNAL CONTAINMENTS. Hillekum, H. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp 307-310 of Reaktortagung, Hannover. Sektion 2. Sicherheit kerntechnischer Anlagen. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978). (In German)

From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).

BETHE-TAIT METHOD;CONTAINMENT;COVER GAS;
EXCURSIONS;PRIMARY COOLANT CIRCUITS;SNR-2
REACTOR;STEELS;SAFETY ENGINEERING

- 3912 KNK-II SAFETY TEST PROGRAMME. Marth, W. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Projektbereich Kompakte Natriumgekuehlte Kernreaktoranlage). pp 294-297 of Reaktortagung, Hannover. Sektion 2. Sicherheit kerntechnischer Anlagen. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978). (In German)
From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).
KNK-2 REACTOR;LMFBR TYPE REACTORS;REACTOR SAFETY;RESEARCH PROGRAMS;SAFETY ENGINEERING

- 3913 COMPARATIVE ANALYSIS OF A HYPOTHETICAL LOSS OF FLOW ACCIDENT IN A LMFBR USING DIFFERENT COMPUTER MODELS FOR A COMMON BENCHMARK PROBLEM. Royl, P.; Maschek, W.; Struwe, D. (Kernforschungszentrum Karlsruhe (Germany, F.R.)); Billington, D.; Perks, M. (UKAEA Reactor Group, Risley); Evrard, A.; Renard, A. (Societe Belge pour l'Industrie Nucleaire, Brussels); DeFremont, R.; Delapparent, D.; Penet, F. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France)). pp 298-302 of Reaktortagung, Hannover. Sektion 2. Sicherheit kerntechnischer Anlagen. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978).
From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).
The comparative loss of flow accident analysis has provided a deeper understanding about possibilities and limits of the different applied approaches to accident analysis. The outlined differences could be explained by more detailed model comparisons. Although the degree of sophistication in modeling was considerably different all codes lead to a reasonably good agreement in integral results such as the molten fuel energy.

- 3914 INFLUENCE OF FISSION GASES ON MECHANICAL ENERGY RELEASE AFTER HYPOTHETICAL COOLANT FLOW ACCIDENTS. Schmuck, P. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). pp 303-306 of Reaktortagung, Hannover. Sektion 2. Sicherheit kerntechnischer Anlagen. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978). (In German)
From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).
BUBBLES;BURNUP;ENERGY YIELD;FISSION PRODUCTS; FLOW BLOCKAGE;GASES;LOSS OF FLOW;PRESSURE RELEASE;QUANTITY RATIO;SNR-1 REACTOR

- 3915 SODIUM BOILING EXPERIMENTS IN NATURAL CONVECTION WITH A THERMAL POWER EQUAL TO THE DECAY HEAT. Straka, M.; Peppier, W.; Kaiser, A. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Reaktorentwicklung). pp 158-161 of Reaktortagung, Hannover. Sektion 1. Reaktorauslegung. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978). (In German)
From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).
AFTER-HEAT REMOVAL;BOILING;FUEL ELEMENT CLUSTERS;NATURAL CONVECTION;SIMULATION;SODIUM; LMFBR TYPE REACTORS

- 3916 CORE RESTRAINT CONCEPT OF THE SNR 300 - POTENTIAL AND RESULTS. Urban, K. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp 837-840 of Reaktortagung, Hannover. Sektion 5. Komponenten und Komponenten-Werkstoffe, Qualitaetssicherung. Eggenstein-Leopoldshafen, Germany, F.R.; Zentralstelle fuer Atomkernenergie-Dokumentation (1978). (In German)
From Reactor congress; Hannover, F.R. Germany (4 Apr 1978).
2D FIAT code. F CODES;FUEL ELEMENTS; MECHANICAL PROPERTIES;PLANNING;REACTOR CORES; SNR-1 REACTOR;SUPPORTS;TRANSIENTS;TWO-DIMENSIONAL CALCULATIONS;REACTOR CORE RESTRAINTS

- 3917 THERMAL ANALYSIS OF LIQUID-METAL FAST BREEDER REACTORS. Tsang, Y.S.; Coffield, R.D. Jr.; Markley, R.A. La Grange Park, IL; American Nuclear Society (1978). 420p. (TID-28732). American Nuclear Society, La Grange Park, IL.
The book contains the following chapters: (1) general description of LMFBR systems, (2) reactor configurations and operating conditions, (3) thermal design criteria and analysis methods, (4) reactor hydrodynamics, (5) steady-state reactor heat transfer, (6) transient reactor heat transfer, and (7) heat exchangers in LMFBR reactors. (DG)

- 3918 (JAPFNR-320) WATER TEST OF THE MONJU PRIMARY SODIUM PUMP MOCK-UP. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). 1978. Translation of Japanese report 8p. Dep. NTIS (US Sales Only), PC A02/MF A01.
Work performed under United States-Japanese Fast Reactor Exchange Program.
This report summarizes the water test of the MONJU primary sodium pump mock-up. The water test was performed in 1977. The cross sectional configuration and general design characteristics of the MONJU primary sodium pump mock-up are given.

- 3919 THOUGHTS ON THE ENERGY SITUATION. Wideroe, R. (Nussbaumen, Baden, Switz). Atomkernenergie; 31: No. 2, 67-73(1978).
The only possible solution to the energy problem in the long term is breeder reactors, according to the author. The non-proliferation treaty (NPT) is explained, and it is shown that supervision of plutonium production necessary for breeder power plants would be possible. The fusion of deuterium and tritium today still remains a "magic talisman" and will only provide power under economically acceptable conditions after a long development time. 27 refs.

- 3920 APPLICATION OF REACTOR SCRAM EXPERIENCE IN RELIABILITY ANALYSIS OF SHUTDOWN SYSTEMS. Edison, G.E.; Gerstner, M.T. Nucl. Saf.; 19: No. 6, 712-722(1978).
Scram experience at a liquid-metal-cooled fast breeder reactor (LMFBR) and 14 commercial light-water reactors (LWRs) has been reviewed and analyzed for application in the reliability analysis of LMFBR shutdown systems. The date and reactor power for each scram were compiled from monthly plant operating reports and personal communications with plant operating personnel. The scram frequency in the Experimental Breeder Reactor II (EBR-II) has been higher than that in commercial LWRs because of its conservative shutdown system

design which leads to more scrams from minor causes. The scram frequency of the EBR-II has declined rapidly with operating experience as some of the overly conservative scrams are eliminated. The EBR-II data trend and other facts suggest that the scram frequency for large LMFBRs is likely to be in the same general range as that for commercial LWRs. The scram frequency curve in LWRs resembles a reliability bathtub curve, with the useful-life phase of operation leveling off at approximately 2.5 scrams per year. A Weibull distribution appears to represent the data well in the early-life portion of the curve. No sign of a wear-out phase is evident after 16 years of operation.

- 3921 **EXPERIMENTAL DETERMINATION OF RELIABILITY DATA.** Brasz, J.; Smit, C.C. (TNO-50MW Test Facility, Hengelo, Netherlands). pp 161-180 of Reliability problems of reactor pressure components. Vol. 1. Vienna; International Atomic Energy Agency (1978). From Symposium on application of reliability techniques to nuclear power plants; Vienna, Austria (10 Oct 1977).
- Failure data of sodium components in the TNO-50MW Test Facility at Hengelo in the Netherlands have been collected and evaluated from start-up in July 1971 up to 1977, covering a total operating time of about 25000h. Failure rates together with the 90% confidence intervals have been determined for the following sodium components: valves, pumps, pipes, pipe welds, thermocouples and trace-heating elements. Statistical data are presented from which the failure rates have been calculated. Different failure modes are considered. The relatively high failure rates for most of the components during the start-up phase of the 50MW Test Facility were followed by much lower constant failure rates. For components for which enough statistics were available, a bath-tub curve could be demonstrated without showing an increase in failure rate due to fatigue or wear. The availability of the 50MW Test Facility increased more than corresponds with the increasing mean time between failure (MTBF) owing to a simultaneous decrease in the mean time to repair (MTTR). The minimum sample size for the experimental determination of a failure rate highly depends on the required confidence level. High confidence levels yield very pessimistic results, especially when the number of failures occurring in the test sample is limited.
- 3922 **PMS: AN EFFECTIVE MANAGEMENT SYSTEM FOR POWER PLANT ENGINEERING DESIGN.** Hover, J.G.; Simmons, R.T. (Burns and Roe, Inc., Oradell, NJ). Am. Soc. Mech. Eng., [Pap.]; No. 78-WA/Mgt-6, 1-11(1978).
- The design and construction of today's large and complex power plants (especially nuclear) demand close control of costs and schedules. Thus, there is an urgent need for an efficient system for controlling cost and schedule performance. This paper describes Burns and Roe's experiences in developing and implementing a management control system that conforms to the requirements of the Department of Energy's Performance Measurement System (PMS). The paper explores the underlying structure of the system and its effectiveness as an important tool for all levels of management of the Clinch River Breeder Reactor Plant Project. Analysis of the PMS criteria issued by the Division of Reactor Research and Technology (RRT) of the Department of Energy (DOE) reveals many basic project elements which are essential to an effective management

control system. Rather than attempt to cover all the criteria, Burns and Roe has chosen to discuss several key elements which we feel can be applied to the engineering and architectural design of any size design or construction project regardless of whether compliance with the PMS is a requirement. These elements are: astute organization of project work; rigorous budget and schedule baseline maintenance; integration of costs, schedules, and work scope; and performance measurement. This paper examines these elements in detail and discusses their application to the Clinch River Project. The paper concludes that the application of the basic management control principals embodied in the criteria, and in particular the four elements discussed, can provide project management and the Client with a highly effective means of managing the detailed costs and complex schedules associated with power plant engineering design.

- 3923 **MAINTENANCE EXPERIENCE WITH SODIUM SYSTEMS AT EXPERIMENTAL BREEDER REACTOR II (EBR-II).** Leman, J.D. (Argonne National Lab., Idaho Falls, ID). Am. Soc. Mech. Eng., [Pap.]; No. 78-JPGC-NE-5, 0-12(1978).
- The reliability of components in sodium and sodium-related systems has been very high over the past fourteen years at EBR-II. Fuel-handling components require the most maintenance, but even these components have functioned with only minor difficulties with over 17,000 operations. One primary pump was removed in 1971 for repairs. The second pump has operated for over 70,000 hr and has not been removed since original start-up. Maintenance of auxiliary systems, such as the purification systems and secondary sodium system, have required the replacement of several bellows sealed valves and several cold traps but very little other repair work has been required. Sodium system component repair requires unique, but usually simple, techniques and equipment. A run-beyond-cladding-breach-program was initiated in the Spring of 1977; in conjunction with this program, some counter measures to control fission-gas release were implemented.
- 3924 **MAINTENANCE AND REQUALIFICATION OF A LARGE LMFBR COMPONENT.** Marshall, J.L.; Coops, W.J.; Baxter, W.F.; Goolsby, D.K. (Hanford Engineering Development Lab., Richland, WA). Am. Soc. Mech. Eng., [Pap.]; No. 78-JPGC-NE-1, 1-13(1978).
- The engineering model of the Fast Flux Test Facility (FFTF) In-Vessel Fuel Handling Machine was utilized to demonstrate the water vapor nitrogen process for cleaning 1 and intricate sodium wetted components, determine criteria for requalification of such components before reinstallation in the FFTF, and test design features of the handling system that will be used at FFTF after the components become radioactive. The paper describes the above activity and the radioactive handling equipment currently being assembled for testing. The paper also describes repair operations of a large radioactive component. 2 refs.
- 3925 **Genehmigungsverfahren fuer Schnelle Brutreaktoren.** Kleine Anfrage an die Bundesregierung. (LICENSING PROCEDURE FOR FAST BREEDERS. QUESTION RAISED IN THE BUNDESTAG). Probst, A.; Lenzer, C.; Pfeifer, A.; Benz, G.; Engelsberger, M.; Gerstein, L.; Hubrig, H.; Riesenhuber, H.; Laufs, P.; Stabenhausen, L. Bonn-Bad Godesberg, Germany. F.R.; Heger (1978). 3p. (In German).
- The Federal government has proposed an

amendment of the Atomic Energy Act according to which breeders can only be licensed if they are facilities with a demonstration and prototype character. The regulation eliminates all reliable, governmental backing conditions for public utilities which want to order breeders. It is not realistic to assume that the power industry will notably invest in the commercial utilization of breeders when the Atomic Energy Act not only does not promise their utilization, but even excludes it for the time being. This also means that there is no reason for considerable governmental financial support for breeders, as governmental promotion only becomes justified when there are prospects of commercial utilization. Furthermore, the question arises whether the reprocessing of spent fuel elements is now endangered, because reprocessing has been defended so far on the ground that by using Pu and U in breeders, uranium can be utilized up to a factor 60 times higher.

to be the identification of options for international cooperation which merited further consideration and which might become the subject of subsequent, more detailed analysis. During the course of the study, modifications in U.S. breeder policy led to an expansion of the analysis to embrace the pros and cons of the major breeder-related policy issues, as well as the respective views of national governments on those issues. The resulting examination of views and patterns of international collaboration emphasizes what was implicit from the outset: Options for international cooperation cannot be fashioned independently of national objectives, policies and programs. Moreover, while similarity of views can stimulate cooperation, this cannot of itself provide compelling justification for cooperative undertakings. Such undertakings are influenced by an array of other national factors, including technological development, industrial infrastructure, economic strength, existing international ties, and historic experience.

- 3926 PERFORMANCE CONSIDERATIONS OF NUCLEAR REACTOR ACTINIDE RECYCLE AS A WASTE MANAGEMENT ALTERNATIVE. Murphy, D.J. Jr. Tucson, AZ; Univ. of Arizona (1978). 161p. University Microfilms Order No. 7812922.

Thesis (Ph. D.).

Transplutonium actinide disposal by fission reactor recycle is examined. The neutronic parameters are evaluated for several average reactor neutron flux energies in the range 325 to 1071 keV. Simplified one-dimensional analytical models of CRBR and EBR-2 in conjunction with the computer codes ANISN and DEER are used for the computations. Results indicate that if the average neutron flux energy is large enough (above approx. 580 keV), then the transplutonium waste actinides behave more as fuels than as wastes. Using the same volumetric power density and radial geometry as EBR-2, a "generic actinide burner reactor" model, fueled solely with transplutonium actinides, is studied. Results show that the actinides are neutronically reactive enough to support a critical chain reaction. The long-term burnup performance indicates that equilibrium is established faster as average flux spectrum energy increases. Increasing the power density increases the time required to establish equilibrium. The specific reactivity (fuel value) and the long-term burnup performance of several actinide-lanthanide mixtures are evaluated. It is shown that although the lanthanides degrade the fuel value of the mixture, the time required to establish equilibrium actinide concentration is essentially unchanged. An analysis of the sensitivity of the results to uncertainties in the actinide nuclear cross section data show that the long-term actinide burnup performance is only slightly affected by relatively large uncertainties in the nuclear area. (DLC)

- 3927 INTERNATIONAL COOPERATION ON BREEDER REACTORS. Gray, J.E.; Kratzer, M.B.; Leslie, K.E.; Paige, H.W.; Shantzis, S.B. New York; Rockefeller Foundation (1978). 180p.

In March 1977, as the result of discussions which began in the fall of 1976, the Rockefeller Foundation requested International Energy Associates Limited (IEAL) to undertake a study of the role of international cooperation in the development and application of the breeder reactor. While there had been considerable international exchange in the development of breeder technology, the existence of at least seven major national breeder development programs raised a *prima facie* issue of the adequacy of international cooperation. The final product of the study was

- 3928 FEASIBILITY OF DECONTAMINATION OF LMFBR COOLANT BY LOW-PRESSURE DISTILLATION. Misra, B.; McPheeters, C.C.; Wolson, R.D. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 676-678(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).

LMFBR TYPE REACTORS; PRIMARY COOLANT CIRCUITS; SODIUM; RADIOACTIVE WASTE PROCESSING; DECONTAMINATION; DISTILLATION

- 3929 PUMA: A TECHNIQUE FOR LMFBR BLANKET FUEL MANAGEMENT ANALYSIS. Lake, J.A. (Westinghouse Advanced Reactors Div., Madison, PA). Trans. Am. Nucl. Soc.; 30: 700-701(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).

LMFBR TYPE REACTORS; BREEDING BLANKETS; FUEL MANAGEMENT; COMPUTER CODES; P CODES; TWO-DIMENSIONAL CALCULATIONS

- 3930 ZPPR ASSEMBLY 9-A 700-MW(E)-SIZED LMFBR BENCHMARK. Olsen, D.N.; Beck, C.L.; Maddison, D.W. (Argonne National Lab., Idaho Falls, ID). Trans. Am. Nucl. Soc.; 30: 703-704(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).

LMFBR TYPE REACTORS; REACTOR KINETICS; BENCHMARKS; MOCKUP; ZPPR REACTOR; CLINCH RIVER BREEDER REACTOR; REACTOR LATTICE PARAMETERS

- 3931 CALCULATIONAL AND EXPERIMENTAL ANALYSIS OF A ^{239}Pu SAMPLE IRRADIATED IN EBR-II. Williams, M.L.; Raman, S. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 30: 706-707(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).

FBR TYPE REACTORS; EBR-2 REACTOR; PLUTONIUM ^{239}Pu ; NEUTRON REACTIONS; REACTOR KINETICS; CROSS SECTIONS; SENSITIVITY; NUCLEAR DATA COLLECTIONS

- 3932 SIMPLE REPRESENTATIONS OF SPECTRUM-AVERAGED CROSS SECTIONS IN LMFBR BLANKETS. Badruzzaman, A.; Wiley, R.; Becker, M. (Rensselaer Polytechnic Inst., Troy, NY). Trans. Am. Nucl. Soc.; 30: 708-709(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).

LMFBR TYPE REACTORS; BREEDING BLANKETS; NEUTRON SPECTRA; CROSS SECTIONS; REACTOR KINETICS

- 3933 COMPARATIVE BREEDING POTENTIAL OF ^{232}Th AND ^{235}U IN LMFBR BLANKETS. Driscoll, M.J.; Medeiros, D.W.; Lancaster, D.B.; Atefi, B. (Massachusetts Inst. of Tech., Cambridge). Trans. Am. Nucl. Soc.; 30: 719-720(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; BREEDING BLANKETS; THORIUM 232; URANIUM 238; CROSS SECTIONS
- 3934 EFFECTS OF THORIUM BREEDER MATERIAL ON SODIUM-VOID REACTIVITY. Beck, C.L.; McFarlane, H.F.; Olsen, D.N. (Argonne National Lab., Idaho Falls). Trans. Am. Nucl. Soc.; 30: 720-722(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; BREEDING BLANKETS; VOID COEFFICIENT; ZPPR REACTOR; CLINCH RIVER BREEDER REACTOR; SODIUM
- 3935 ZPPR ASSEMBLY 8: A STUDY OF THORIUM BLANKET ZONES IN A HETEROGENEOUS LMFBR. Ducat, G.A.; Beck, C.L.; Gasidio, J.M.; Olsen, D.N.; Kaiser, R.E.; Grasseschi, G.L. (Argonne National Lab., Idaho Falls, IL). Trans. Am. Nucl. Soc.; 30: 722-723(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; ZPPR REACTOR; CLINCH RIVER BREEDER REACTOR; MOCKUP; BREEDING BLANKETS; THORIUM
- 3936 WIRE WRAP-CLADDING MECHANICAL INTERACTION IN SODIUM-BOUNDED ELEMENTS. Kerrisk, J.F.; Petty, R.L.; DeMuth, N.S.; Latimer, T.W. (Los Alamos Scientific Lab., NM). Trans. Am. Nucl. Soc.; 30: 186-186(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; FUEL PINS; FUEL CANS; DEFORMATION; FUEL-CLADDING INTERACTIONS
- 3937 PULSED MAGNETIC WELDING OF BREEDER REACTOR FUEL PIN END CLOSURES. Brown, W.F. (Hanford Engineering Development Lab., Richland, WA); Bandas, J.; Olson, N.T. Trans. Am. Nucl. Soc.; 30: 186-187(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; FUEL PINS; FABRICATION; WELDED JOINTS; STAINLESS STEEL-316
- 3938 RADIONUCLIDE TRAP FOR LIQUID-METAL-COOLED REACTORS. McGuire, J.C.; Brehm, W.F. (Hanford Engineering Development Lab., Richland, WA). Trans. Am. Nucl. Soc.; 30: 189-190(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; PRIMARY COOLANT CIRCUITS; CORROSION PRODUCTS; TRAPS; GETTERS; SPECIFICATIONS
- 3939 EVAPORATIVE REMOVAL OF SODIUM FROM REACTOR COMPONENTS. Welch, F.H.; Steele, O.P. III. (Rockwell International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 30: 190-191(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR COMPONENTS; SODIUM; SURFACE CLEANING
- 3940 DEVELOPMENT OF THE GLYCOLIC--CITRIC ACIDS (GCA) PROCESS FOR DECONTAMINATION OF LMFBR COMPONENTS. Hill, E.F. (Rockwell International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 30: 191-192(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; REACTOR COMPONENTS; SURFACE CLEANING; DECONTAMINATION; GLYCOLIC ACID; CITRIC ACID
- 3941 DUCT DILATION EXPERIMENTS IN EBR-II. Moran, T.J.; Flinn, J.E. (Argonne National Lab., Idaho Falls, ID). Trans. Am. Nucl. Soc.; 30: 192-193(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR INTERNALS; REACTOR COMPONENTS; DEFORMATION; CREEP; SWELLING; PHYSICAL RADIATION EFFECTS; STAINLESS STEEL-304
- 3942 EFFECTS OF MULTIPLE TRANSIENTS ON FAST REACTOR FUEL PIN CLADDING MECHANICAL PROPERTIES. Duncan, D.R.; Johnson, G.D.; Hunter, C.W. (Hanford Engineering Development Lab., Richland, WA). Trans. Am. Nucl. Soc.; 30: 196-198(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; FUEL PINS; FUEL CANS; TRANSIENTS; THERMAL STRESSES; STAINLESS STEEL-316; PHYSICAL RADIATION EFFECTS; FRACTURE PROPERTIES; FAILURES
- 3943 ASSUMED-MODE MODELING FOR FAST REACTOR CORE SEISMIC ANALYSIS. Moran, T.J. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 213-214(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; REACTOR CORES; SEISMIC EFFECTS; MATHEMATICAL MODELS; SODIUM; HYDRAULICS; STRESSES
- 3944 MODELING FLUID STRUCTURE COUPLING FOR LMFBR CORES. Moran, T.J.; Yang, C.I. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 214-215(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; REACTOR CORES; SEISMIC EFFECTS; HYDRAULICS; SODIUM; STRESSES; MATHEMATICAL MODELS
- 3945 MONTE CARLO BENCHMARKING OF FAST REACTOR COMPUTATIONS. Gelbard, E.M.; Prael, R.E.; Reynolds, B.G. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 227(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; REACTOR KINETICS; MONTE CARLO METHOD
- 3946 CORE STRUCTURAL ANALYSIS CODES VERIFICATION/QUALIFICATION. McLennan, G.A.; Moran, T.J.; Cha, B.K. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 228-229(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR CORES; FUEL ASSEMBLIES; DEFORMATION; COMPUTER CODES

- 3947 FUEL FAILURE MONITORING SYSTEM FOR CRBRP. Keshishian, V. (Atomics International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 30: 504-505(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
CLINCH RIVER BREEDER REACTOR; REACTOR INSTRUMENTATION; FAILED ELEMENT DETECTION
- 3948 CALIBRATION OF ELECTROMAGNETIC FLOWMETERS BY AVERAGE-FREQUENCY METHOD. Fujino, M.; Izumi, S. (Hitachi Ltd., Ibaraki, JP); Fukuda, T.; Sano, K. Trans. Am. Nucl. Soc.; 30: 510-512(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; PRIMARY COOLANT CIRCUITS; FLOWMETERS; CALIBRATION
- 3949 DEVELOPMENT OF A STATE-OF-ART THERMAL MIXER. Kasza, K.E.; Bobis, J.P. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 524-525(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
CLINCH RIVER BREEDER REACTOR; PRIMARY COOLANT CIRCUITS; SODIUM; LIQUID FLOW; PIPES; MIXERS; PERFORMANCE; THERMAL STRESSES; FAILURES
- 3950 COMPARISONS OF XX08 STEADY-STATE TEMPERATURE MEASUREMENTS WITH ANALYTICAL PREDICTIONS. Betten, P.R.; Chang, L.K.; Tokar, J.V. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 528-529(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; FUEL ASSEMBLIES; HYDRAULICS; HEAT TRANSFER; COMPUTER CODES
- 3951 TRIANGULAR EDGE-SUBCHANNEL WIRE-WRAP PHIFUNCTIONS FOR FLOW BLOCKAGE ANALYSIS. Lin, E.-I.H.; Sha, W.T. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 529-531(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; FUEL ASSEMBLIES; FLOW BLOCKAGE; COMPUTER CODES; HYDRAULICS; TEMPERATURE GRADIENTS
- 3952 THERMAL CONDUCTIVITY DURING INITIAL STAGE RESTRUCTURING IN SPHERE-PAC FUEL. Ades, M.; Peddicord, K.L.; Guenther, R.J. (Oregon State Univ., Corvallis). Trans. Am. Nucl. Soc.; 30: 531-532(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; FUEL PARTICLES; THERMAL CONDUCTIVITY
- 3953 THREE-DIMENSIONAL NUMERICAL SIMULATION OF FLOW BLOCKAGE IN A 19-PIN LMFBR FUEL ASSEMBLY. Domanus, H.M.; Sha, W.T. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 534-535(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; FUEL PINS; FUEL ASSEMBLIES; FLOW BLOCKAGE; HYDRAULICS; HEAT TRANSFER
- 3954 THREE-DIMENSIONAL COUPLED COOLANT CELL HEAT TRANSFER ANALYSIS OF FINITE HEXAGONAL LMFBR BUNDLES. Wong, C.N.; Wolf, L. (Massachusetts Inst. of Tech., Cambridge). Trans. Am. Nucl. Soc.; 30: 540-541(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; FUEL ASSEMBLIES; HEAT TRANSFER; THREE-DIMENSIONAL CALCULATIONS; TEMPERATURE GRADIENTS
- 3955 PRESSURE DROP MEASUREMENTS IN LMFBR WIRE-WRAPPED BLANKET BUNDLES. Chiu, C.; Todreas, N.E.; Rohsenow, W.M. (Massachusetts Inst. of Tech., Cambridge). Trans. Am. Nucl. Soc.; 30: 541-543(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; FUEL ASSEMBLIES; PRESSURE DROP; HYDRAULICS; CONFIGURATION
- 3956 TURBULENT MIXING EXPERIMENT AND MODEL FOR WIRE-WRAPPED ASSEMBLIES. Chiu, C.; Todreas, N.E.; Rohsenow, W.M. (Massachusetts Inst. of Tech., Cambridge). Trans. Am. Nucl. Soc.; 30: 547-548(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; FUEL RODS; HYDRAULICS; TURBULENT FLOW; FUEL ASSEMBLIES
- 3957 CRITICAL HEAT FLUX MEASUREMENTS IN AN LMFBR EVAPORATOR TUBE. France, D.M.; Chiang, T.; Carlson, R.D. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 548(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; STEAM GENERATORS; CRITICAL HEAT FLUX
- 3958 POOL-TYPE LARGE BREEDER REACTOR PLANT DESIGN. McDonald, J.; Brunings, J.; Moody, E. (Atomics International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 30: 564-565(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; SPECIFICATIONS
- 3959 1000-MW(E) POOL PLANT LMFBR FABRICATION AND CONSTRUCTION. Moody, E. (Atomics International, Canoga Park, CA); Cook, J.P.; Gill, C.J. Trans. Am. Nucl. Soc.; 30: 565-566(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; FABRICATION; CONSTRUCTION
- 3960 IMPACT OF DESIGN CONSTRAINTS ON THE SELECTION OF FEASIBLE DESIGNS. Lam, P.S.K.; Turski, R.B.; Barthold, W.P. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 573-575(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; FUEL PINS; DESIGN; COMPARATIVE EVALUATIONS
- 3961 GENERIC TRANSIENT LIMIT CURVES FOR LMFBR CORE DESIGN GUIDANCE. Coffield, R.D. Jr.; Markley, R.A.; Dickson, P.W.; Anderson, H.C. III. (Westinghouse Advanced Reactors Div., Madison, PA). Trans. Am. Nucl. Soc.; 30: 575-576(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; REACTOR CORES; FUEL PINS; SPECIFICATIONS; TRANSIENTS; DESIGN; PERFORMANCE

- 3962 NUCLEAR DESIGN OF HETEROGENEOUS CARBIDE-FUELED LMFBR CORES. Brittingham, J.C.; Vitt, J.A. (Atomics International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 30: 576-578(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; REACTOR CORES; SPECIFICATIONS
- 3963 PHYSICS PERFORMANCE OF HETEROGENEOUS LMFBRs WITH INTERNAL AXIAL BLANKETS. Su, S.F. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 578-579(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; REACTOR CORES; REACTOR KINETICS; REACTIVITY WORTHS; REACTIVITY COEFFICIENTS; CONFIGURATION; SPECIFICATIONS
- 3964 SELECTION OF REACTOR COVER SEALS AND BEARINGS FOR A 1000-MW(E) LMFBR. McEdwards, J.A.; Brunings, J.E.; Materas, J.L.; Sayles, C.W. (Atomics International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 30: 579-580(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; REACTOR VESSELS; CLOSURES; SEALS; SPECIFICATIONS
- 3965 DECOMMISSIONING OF FERMI-1. Alexanderson, E.L. (Detroit Edison Co., Detroit, MI). Trans. Am. Nucl. Soc.; 30: 551-553(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
ENRICO FERMI-1 REACTOR; REACTOR DECOMMISSIONING
- 3966 THERMAL AND ECONOMIC PERFORMANCE OF HETEROGENEOUS CARBIDE-FUELED REACTORS. Felten, L.D.; Galluzzo, N.G.; Vitt, J.A.; von Arx, A.V. (Atomics International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 30: 487-488(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; ECONOMICS; REACTOR OPERATION; FUEL CYCLE; PERFORMANCE; PLUTONIUM CARBIDES; URANIUM CARBIDES; THORIUM CARBIDES
- 3967 B₄C ABSORBER PIN IN-REACTOR RUPTURE TEST. Pitner, A.L.; Hollenberg, G.W.; Jackson, J.L.; Basmajian, J.A. (Hanford Engineering Development Lab., Richland, WA). Trans. Am. Nucl. Soc.; 30: 188-189(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
BORON CARBIDES; CONTROL ELEMENTS; RUPTURES; TESTING; EBR-2 REACTOR; PRESSURE DEPENDENCE; PHYSICAL RADIATION EFFECTS
- 3968 IN-CORE GAS RELEASE CAPSULE DEVELOPMENT. Kolowith, R.; Miller, W.C. (Hanford Engineering Development Lab., Richland, WA). Trans. Am. Nucl. Soc.; 30: 635-636(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
FFTF REACTOR; FAILED ELEMENT MONITORS; TESTING; FISSION PRODUCTS; GASES
- 3969 CORRELATIONS OF CALCULATED REACTIVITIES OF EBR-II LOADINGS. Meneghetti, D.; Opelka, J.H. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 707-708(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
EBR-2 REACTOR; REACTIVITY; REACTOR CORES; CONFIGURATION
- 3970 EXPERIENCE IN REMOTELY FABRICATING EBR-II FUEL. Feldman, M.J. (Oak Ridge National Lab., TN); Grant, N.R.; Hampson, D.C.; Fryer, R.M. Trans. Am. Nucl. Soc.; 30: 309-310(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
EBR-2 REACTOR; FUEL ELEMENTS; FABRICATION
- 3971 INSPECTION TECHNIQUES AND PROCESSES FOR CONTROLLING FFTF FUEL QUALITY. Powers, H.G.; McLemore, D.R. (Hanford Engineering Development Lab., Richland, WA). Trans. Am. Nucl. Soc.; 30: 324-325(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
FFTF REACTOR; FUEL PELLETS; FUEL PINS; INSPECTION; QUALITY CONTROL
- 3972 MECHANICAL PROPERTIES OF TRANSIENT-TESTED IRRADIATED FAST REACTOR CLADDING. Johnson, G.D.; Hunter, C.W. (Hanford Engineering Development Lab., Richland, WA). Trans. Am. Nucl. Soc.; 30: 195-196(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
FFTF REACTOR; FUEL CANS; SWELLING; STRESSES; TRANSIENTS; PHYSICAL RADIATION EFFECTS; STAINLESS STEEL-316
- 3973 COMPARATIVE STUDIES OF FFTF UPPER-PLENUM MIXING AND STRATIFICATION. Yang, J.W. (Brookhaven National Lab., Upton, NY). Trans. Am. Nucl. Soc.; 30: 535-536(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
FFTF REACTOR; REACTOR CORES; PRIMARY COOLANT CIRCUITS; HYDRAULICS; HEAT TRANSFER; MIXING
- 3974 FAST FLUX TEST FACILITY CONTAINMENT BUILDING INTEGRATED LEAK-RATE TEST. Farabee, O.A. (Dept. of Energy, Richland, WA); James, E.B. Trans. Am. Nucl. Soc.; 30: 634-635(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
FFTF REACTOR; CONTAINMENT BUILDINGS; LEAK TESTING; CONTROLLED ATMOSPHERES
- 3975 IMPROVED METHOD FOR CONDUCTING FAILED FUEL EXPERIMENTS IN LWRs. Thomas, R.W.; Klingler, W.B. (EG and G Idaho, Idaho Falls). Trans. Am. Nucl. Soc.; 30: 637-638(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
ETR sodium loop safety facility. LMFBR TYPE REACTORS; FUEL ELEMENT FAILURE; TRANSIENTS; LOSS OF FLOW; TESTING; ETR REACTOR
- 3976 MEASUREMENT OF INTEGRAL PHYSICS PARAMETERS IN SIMULATED LMFBR MELTDOWN CONFIGURATIONS. Bhattacharyya, S.K.; Bucher, R.G.; Morman, J.A.; Smith, D.M.; Bennett, E.F.; Cohn, C.E.; Robinson, W.R. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 699-

- 700(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; MELTDOWN; REACTIVITY WORTHS; MULTIPLICATION FACTORS; CONTROL ROD WORTHS; CONFIGURATION
- 3977 LOCATION-DEPENDENT COMMON-CAUSE ANALYSIS WITH AN APPLICATION TO FIRES. Leaver, D.; vonHerrmann, J.; Olmos, J.; Boardman, R. (Science Applications, Inc., Palo Alto, CA). Trans. Am. Nucl. Soc.; 30: 360-362(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
CLINCH RIVER BREEDER REACTOR; REACTOR SAFETY; FIRES; FIRE HAZARDS; RISK ASSESSMENT; MATHEMATICAL MODELS
- 3978 EXPERIMENTAL DETERMINATION OF DECAY POWER IN EBR-II. Gillette, J.L.; Tokar, J.V.; Chang, L.K.; Mohr, D. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 412(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR SHUTDOWN; AFTER-HEAT; HEAT TRANSFER; HYDRAULICS; EBR-2 REACTOR
- 3979 LMFBR TRANSITION TO NATURAL CIRCULATION FROM AN INITIAL ISOTHERMAL STATE. Singer, R.M.; Gillette, J.L.; Mohr, D.; Tokar, J.V. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 412-413(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; LOSS OF FLOW; REACTOR SAFETY; HYDRAULICS; HEAT TRANSFER; NATURAL CONVECTION; AFTER-HEAT REMOVAL
- 3980 SODIUM VOID REACTIVITY REDUCTION IN METAL-FUELED LMFBRs. Orehwa, Y.; Barthold, W.P. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 415-416(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; REACTOR SAFETY; LOSS OF FLOW; SODIUM; VOID COEFFICIENT; REACTIVITY WORTHS
- 3981 RADIATION HEAT TRANSFER IN A 19-PIN SODIUM-VOIDED BUNDLE. Machbitz, M. (Massachusetts Inst. of Tech., Cambridge); Budiman, B.; Roberts, Y.Y.; Senkan, S.M.; Han, J.T. Trans. Am. Nucl. Soc.; 30: 416-417(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; FLOW BLOCKAGE; FUEL ASSEMBLIES; HEAT TRANSFER; AFTER-HEAT; FUEL PINS
- 3982 MAXIMUM SODIUM TEMPERATURE CORRELATION FOR SIX-CHANNEL FBR SUBASSEMBLY BLOCKAGES. Hanus, N.; Han, J.T.; Fontana, M.H.; Wantland, J.L. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 30: 417-419(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; REACTOR SAFETY; FUEL ASSEMBLIES; FUEL PINS; SODIUM; BOILING; HEAT TRANSFER; TEMPERATURE GRADIENTS; FLOW BLOCKAGE
- 3983 POTENTIAL GAS ENTRY INTO FFTF AFTER A POSTULATED PIPE RUPTURE. McCall, T.B. (Westinghouse Electric Corp., Richland, WA); Markowski, F.J.; Shire, P.R. Trans. Am. Nucl. Soc.; 30: 419-420(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
FFTF REACTOR; REACTOR SAFETY; LOSS OF COOLANT; PRIMARY COOLANT CIRCUITS; PIPES; FAILURES; HEAT TRANSFER; HYDRAULICS
- 3984 COMPARISON OF THE BEHAVIOR DURING A TUC OF CRBR CARBIDE RELOAD DESIGNS TO THE REFERENCE OXIDE DESIGN. Bergeron, R.J. (Combustion Engineering, Inc., Windsor, CT). Trans. Am. Nucl. Soc.; 30: 420-421(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
CLINCH RIVER BREEDER REACTOR; REACTOR CORES; REACTOR SAFETY; FUEL PINS; TRANSIENTS; REACTIVITY WORTHS; MOTION
- 3985 BEHAVE-SST INTERPRETATION OF TREAT EXPERIMENTS HUT 5-5A AND HUT 5-5B. Gregoire, K.E.; Atcheson, D.B. (General Electric Co., Sunnyvale, CA). Trans. Am. Nucl. Soc.; 30: 422-423(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; TRANSIENTS; REACTOR SAFETY; FUEL CANS; FUEL ELEMENT FAILURE
- 3986 STRAIN RATE EFFECTS IN TRANSIENT OVERPOWER CLADDING FAILURE ANALYSES. Atcheson, D.B.; Cole, C.V. (General Electric Co., Sunnyvale, CA). Trans. Am. Nucl. Soc.; 30: 423-424(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; FUEL CANS; STRAINS; FUEL ELEMENT FAILURE; TRANSIENTS; STAINLESS STEEL-316; COMPUTER CODES
- 3987 ANALYSIS OF A HYPOTHETICAL LMFBR ACCIDENT USING THE CODES FUTURE AND FRUMP. Grosberg, A.J. (Imperial Coll. of Science and Tech., London, England); Head, J.L.; Tregonning, K. Trans. Am. Nucl. Soc.; 30: 424-426(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; FUEL CANS; STRAINS; DEFORMATION; REACTOR SAFETY; TRANSIENTS; REACTOR ACCIDENTS; COMPUTER CODES; FUEL ELEMENT FAILURE
- 3988 DYNAMIC INSTABILITY ANALYSIS OF RUPTURE DISKS BY FINITE ELEMENT METHOD. Hsieh, B.J. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 426-427(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; STEAM GENERATORS; PRESSURE RELEASE; ENGINEERED SAFETY SYSTEMS; PERFORMANCE; RELIEF VALVES; FINITE ELEMENT METHOD
- 3989 THREE-DIMENSIONAL FLUID-STRUCTURE INTERACTION CALCULATIONS FOR THE DYNAMIC RESPONSE OF A POOL-TYPE REACTOR'S IN-TANK COMPONENT. Kulak, R.F.; Fiala, C. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 428-429(1978). (CONF-7811109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR INTERNALS; REACTOR CORE DISRUPTION; REACTOR SAFETY; HYDRAULICS; THREE-DIMENSIONAL CALCULATIONS
- 3990 SODIUM SLUG KINETIC ENERGY BEHAVIOR FOR

- VARYING HCDA ENERGETICS. Zeuch, W.R.; Bratis, J.C. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 431-432(1978). (CONF-781109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR CORE DISRUPTION; HYDRAULICS; SODIUM; LIQUID FLOW; STRESSES
- 3991 ANALYSIS OF PRESTRESSED CONCRETE VESSELS FOR HCDA LOADS. Marchertas, A.H. (Argonne National Lab., IL); Belytschko, T.B. Trans. Am. Nucl. Soc.; 30: 432-424(1978). (CONF-781109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR CORE DISRUPTION; REACTOR VESSELS; PRESTRESSED CONCRETE; STRESS ANALYSIS; STRESSES
- 3992 MECHANICAL FAILURE OF SUBASSEMBLY DUCTS AND ITS EFFECTS ON TRANSITION-PHASE ANALYSIS. Hakim, S.J.; Kennedy, J.M. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 433-435(1978). (CONF-781109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
FTTF REACTOR; REACTOR SAFETY; LOSS OF FLOW; FUEL PINS; FUEL CANS; FUEL ELEMENT FAILURE; COMPUTER CODES; MELTDOWN
- 3993 DEBRIS BED AND SACRIFICIAL MATERIALS INTERACTIONS AT HIGH TEMPERATURES. Plein, H.G.; Carlson, G.A. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 30: 436-437(1978). (CONF-781109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR CORE DISRUPTION; MELTDOWN; CORIUM; URANIUM DIOXIDE; STAINLESS STEELS; MAGNESIUM OXIDES; THORIUM OXIDES; COMPATIBILITY; CHEMICAL REACTIONS; HYDRAULICS
- 3994 INTERPRETATION OF DEBRIS BED DRYOUT OBSERVED DURING THE D-3 EXPERIMENT. Rivard, J.B. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 30: 437-438(1978). (CONF-781109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR CORE DISRUPTION; MELTDOWN; CORIUM; URANIUM DIOXIDE; DRYOUT; HEAT TRANSFER
- 3995 FUEL DISASSEMBLY EXPERIMENTS USING THE CAPACITOR DISCHARGE VAPORIZATION (CDV) TECHNIQUE. Wright, A.L.; White, J.S.; Kress, T.S.; Bertini, H.W. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 30: 438-439(1978). (CONF-781109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR CORE DISRUPTION; FUEL PELLETS; URANIUM DIOXIDE; RADIOACTIVE AEROSOLS; DECOMPOSITION; EVAPORATION
- 3996 MODELING EJECTION OF FRESH FUEL INTO SODIUM UNDER PROMPT-BURST CONDITIONS. Garner, P.L. (Argonne National Lab., IL); Abramson, P.B.; Pizzica, P.A.; Portugal, J.L.; Young, M.F. Trans. Am. Nucl. Soc.; 30: 439-440(1978). (CONF-781109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR CORE DISRUPTION; FUEL-COOLANT INTERACTIONS; COMPUTER CODES; HYDRAULICS
- 3997 EXPLORATORY STUDIES ON CARBIDE FUEL/COOLANT INTERACTIONS. Mync, L. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 441(1978). (CONF-781109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR CORE DISRUPTION; FUEL-COOLANT INTERACTIONS; URANIUM CARBIDES; PLUTONIUM CARBIDES; COMPUTER CODES
- 3998 CAMEL SEVEN-PIN FUEL SWEEPOUT TESTS. Spencer, B.W.; Bova, L.; Goldfuss, G.T.; Henry, R.E.; Raridon, D. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 443-444(1978). (CONF-781109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; TRANSIENTS; REACTOR CORE DISRUPTION; CORIUM; HYDRAULICS; LIQUID FLOW; FUEL PINS; FUEL ELEMENT FAILURE
- 3999 SYSTEM HYDRAULIC EFFECTS ON FUEL SWEEPOUT IN THE HYPOTHETICAL TOP ACCIDENT. Spencer, B.W. (Argonne National Lab., IL); Henry, R.E.; Jones, B.G. Trans. Am. Nucl. Soc.; 30: 444-445(1978). (CONF-781109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; TRANSIENTS; REACTOR CORE DISRUPTION; CORIUM; HYDRAULICS; LIQUID FLOW; FUEL PINS; FUEL ELEMENT FAILURE
- 4000 RESULTS OF FUEL FREEZING TESTS WITH SIMULATED CRBR-TYPE FUEL PINS. Spencer, B.W.; Roth, R.L.; Goldfuss, G.T.; Henry, R.E. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 446-447(1978). (CONF-781109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
CLINCH RIVER BREEDER REACTOR; REACTOR SAFETY; FUEL PINS; LOSS OF FLOW; TRANSIENTS; FUEL ELEMENT FAILURE; URANIUM DIOXIDE; LIQUID FLOW; SOLIDIFICATION; CORIUM
- 4001 MECHANISTIC MODEL OF FUEL FLOW REGIMES AND FUEL PLATEOUT IN LMFBR OVERPOWER ACCIDENTS. Tentner, A.M.; Wider, H.U.; Bowers, C.H. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 448-450(1978). (CONF-781109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; REACTOR SAFETY; TRANSIENTS; REACTOR CORE DISRUPTION; CORIUM; SOLIDIFICATION; LIQUID FLOW
- 4002 TRANSIENT FUEL SOLIDIFICATION AND RELOCATION FROM CORE DURING TRANSITION-PHASE CONDITIONS. Greene, G.A. (Brookhaven National Lab., Upton, NY); Jones, O.C. Jr.; Leonhardt, W.J.; Kazimi, M.S. Trans. Am. Nucl. Soc.; 30: 451-452(1978). (CONF-781109--).
From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; MELTDOWN; LIQUID FLOW; HYDRAULICS; CORIUM; SOLIDIFICATION
- 4003 BOILING GLOW REGIMES FOR CARBIDE FUEL AND STEEL. Henry, T.P.; Siegmann, E.R. (Combustion Engineering, Inc., Windsor, CT).

- Trans. Am. Nucl. Soc.; 30: 452-453(1978). (CONF-7811109--).
- From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY;
REACTOR CORE DISRUPTION; LIQUID FLOW; BOILING;
HYDRAULICS; CORIUM; URANIUM DIOXIDE; URANIUM
CARBIDES; STAINLESS STEELS; HEAT TRANSFER
- 4004 ATTENUATION OF HCDA WORK POTENTIAL BY
HEAT TRANSFER. Chan, S.H. (Univ. of
Wisconsin, Milwaukee, WI); Tsai, C.F.; Cho, D.
Trans. Am. Nucl. Soc.; 30: 453-455(1978).
(CONF-7811109--).
- From 1978 winter meeting of American Nuclear
Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY;
REACTOR CORE DISRUPTION; HEAT TRANSFER
- 4005 CONDENSATION FROM A LARGE HCDA VAPOR-GAS
BUBBLE ONTO STRUCTURES. Ozisik, M.N. (North
Carolina State Univ., Raleigh, NC); Kress, T.S.;
White, J.S. Trans. Am. Nucl. Soc.; 30: 455-
456(1978). (CONF-7811109--).
- From 1978 winter meeting of American Nuclear
Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY;
REACTOR CORE DISRUPTION; EVAPORATION; CORIUM;
VAPOR CONDENSATION
- 4006 EFFECTS OF SODIUM ENTRAINMENT AND HEAT
TRANSFER WITH UO_2 VAPOR DURING AN HCDA.
Corradini, M.; Rohsenow, W.M.; Todreas, N.E.
(Massachusetts Inst. of Tech., Cambridge).
Trans. Am. Nucl. Soc.; 30: 456-457(1978).
(CONF-7811109--).
- From 1978 winter meeting of American Nuclear
Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY;
REACTOR CORE DISRUPTION; FUEL-COOLANT
INTERACTIONS; URANIUM DIOXIDE; TWO-PHASE FLOW;
HEAT TRANSFER
- 4007 BOILUP THRESHOLD FOR THE BOTTLED-UP
TRANSITION-PHASE POOL. Martin, F.J.
(Westinghouse Electric Corp., Richland, WA).
Trans. Am. Nucl. Soc.; 30: 457-458(1978).
(CONF-7811109--).
- From 1978 winter meeting of American Nuclear
Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY;
REACTOR CORE DISRUPTION; HYDRAULICS; BOILING; TWO-
PHASE FLOW
- 4008 ACOUSTIC DATA ANALYSIS: TREAT TEST R12.
Koopman, R.N.; Carey, W.M. (Argonne National
Lab., IL). Trans. Am. Nucl. Soc.; 30: 461-
462(1978). (CONF-7811109--).
- From 1978 winter meeting of American Nuclear
Society; Washington, DC, USA (12 Nov 1978).
FFTF REACTOR; FUEL PINS; TRANSIENTS; REACTOR
SAFETY; TREAT REACTOR; FUEL ELEMENT FAILURE;
FAILED ELEMENT DETECTION
- 4009 STATISTICAL MODEL TO IDENTIFY BOILING
INCEPTION IN TREAT TESTS. Doerner, R.C.
(Argonne National Lab., IL); Meek, C.C.; Hurt,
R.F. Trans. Am. Nucl. Soc.; 30: 462-
463(1978). (CONF-7811109--).
- From 1978 winter meeting of American Nuclear
Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY;
LOSS OF FLOW; BOILING; SODIUM; HEAT TRANSFER;
BOILING DETECTION; TWO-PHASE FLOW
- 4010 STATISTICAL ANALYSIS OF 17 TREAT
EXPERIMENTS. Burns, R.D. III.; Scott, J.H.
(Los Alamos Scientific Lab., NM). Trans. Am.
Nucl. Soc.; 30: 463-465(1978). (CONF-7811109-
--).
- From 1978 winter meeting of American Nuclear
Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; FUEL ELEMENT
FAILURE; MELTDOWN; FAILED ELEMENT DETECTION;
STATISTICS; TREAT REACTOR; REACTOR SAFETY
- 4011 APPLICATION OF A NONREDUNDANT PINHOLE
ARRAY TO A TEST FUEL ELEMENT AT TREAT. Han,
K.S.; Berzins, G.J. (Los Alamos Scientific
Lab., NM). Trans. Am. Nucl. Soc.; 30: 466-
467(1978). (CONF-7811109--).
- From 1978 winter meeting of American Nuclear
Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY;
REACTOR CORE DISRUPTION; FUEL PINS; MOTION; TREAT
REACTOR; TEST FACILITIES
- 4012 MEASUREMENT OF THE SOUND PRODUCED DURING
STEAM INJECTIONS INTO SODIUM. Claytor, T.N.
(Argonne National Lab., IL); Carey, W.M.
Trans. Am. Nucl. Soc.; 30: 468-469(1978).
(CONF-7811109--).
- From 1978 winter meeting of American Nuclear
Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY;
MOLTEN METAL-WATER REACTIONS; SODIUM; STEAM;
GENERATORS; LEAKS; SIMULATION; TEST FACILITIES;
ACOUSTIC MONITORING
- 4013 EFFECT OF RADIATIVE HEAT TRANSFER ON THE
THERMAL-FLUID BEHAVIOR OF MOLTEN UO_2 . Nguyen,
D.H.; Plein, H.G. (Sandia Labs., Albuquerque,
NM). Trans. Am. Nucl. Soc.; 30: 470-
471(1978). (CONF-7811109--).
- From 1978 winter meeting of American Nuclear
Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY;
MELTDOWN; CORIUM; HEAT TRANSFER; HYDRAULICS;
URANIUM DIOXIDE; AFTER-HEAT REMOVAL
- 4014 DEBRIS TRANSPORT IN HORIZONTAL PIPING.
Brown, R.G. (Westinghouse Advanced Reactors
div., Madison, PA). Trans. Am. Nucl. Soc.; 30:
471-472(1978). (CONF-7811109--).
- From 1978 winter meeting of American Nuclear
Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY;
REACTOR CORE DISRUPTION; MELTDOWN; CORIUM; HEAT
TRANSFER; AFTER-HEAT REMOVAL; HYDRAULICS; PRIMARY
COOLANT CIRCUITS; PIPES
- 4015 NATURAL-CONVECTION HEAT TRANSFER IN A
BED OF FUEL DEBRIS COOLED AT BOTH TOP AND
BOTTOM. Cherng, J.C.; Dhir, V.K.; Catton, I.
(Univ. of California, Los Angeles). Trans.
Am. Nucl. Soc.; 30: 472-473(1978). (CONF-
7811109--).
- From 1978 winter meeting of American Nuclear
Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY;
MELTDOWN; CORIUM; AFTER-HEAT REMOVAL; HEAT
TRANSFER
- 4016 RADIATIVE HEAT TRANSFER IN A HORIZONTAL
MOLTEN FUEL LAYER WITH VOLUMETRIC HEATING.
Cho, D.H. (Argonne National Lab., IL); Chan,
S.H.; Hauser, G.M. Trans. Am. Nucl. Soc.; 30:
473-474(1978). (CONF-7811109--).
- From 1978 winter meeting of American Nuclear
Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY;
MELTDOWN; AFTER-HEAT REMOVAL; CORIUM; HEAT
TRANSFER; URANIUM DIOXIDE; NATURAL CONVECTION

- 4017 TIME-RESOLVED PENETRATION MEASUREMENTS OF LIQUID SODIUM INTO LIMESTONE CONCRETE. Smaardyk, J.E.; Sutherland, H.J.; Kent, L.A. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 30: 474-476(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR ACCIDENTS; CONCRETES; SODIUM; CHEMICAL REACTIONS; REACTOR MATERIALS

- 4018 LMFBR CONTAINMENT PERFORMANCE STUDIES WITH A SIMPLE COMPUTER CODE. Booth, D.L. (Nuclear Installations Inspectorate, London, England). Trans. Am. Nucl. Soc.; 30: 476-477(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR ACCIDENTS; CONTAINMENT BUILDINGS; SODIUM; VAPORS; EVAPORATION; STRESSES; COMPUTER CODES; PRESSURE GRADIENTS

- 4019 GRAVITATIONAL COLLISION EFFICIENCY OF LMFBR AEROSOLS WITH SIZE-DEPENDENT DENSITY. Pertner, G.A.; Loyalka, S.K. (Univ. of Missouri, Columbia). Trans. Am. Nucl. Soc.; 30: 477-478(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR CORE DISRUPTION; AEROSOLS; MOTION; COMPUTER CODES

- 4020 NEW SCHEME FOR SOLVING THE COBRA ENERGY EQUATIONS DURING NATURAL-CIRCULATION CONDITIONS. Masterson, R.E.; Wheeler, C.L. (Battelle Northwest Labs., Richland, WA). Trans. Am. Nucl. Soc.; 30: 204-205(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; LOSS OF FLOW; REACTOR SAFETY; FUEL ASSEMBLIES; NATURAL CONVECTION; HEAT TRANSFER; HYDRAULICS; COMPUTER CODES

- 4021 EXPERIMENTAL OBSERVATIONS OF INTERSUBASSEMBLY HEAT TRANSFER EFFECTS DURING NATURAL CIRCULATION. Gillette, J.L.; Tokar, J.V.; Singer, R.M. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 30: 537-538(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; LOSS OF FLOW; AFTER-HEAT; FUEL ASSEMBLIES; HEAT TRANSFER; NATURAL CONVECTION

- 4022 DEPARTMENT OF ENERGY AUTHORIZATION. VOLUME IV. HEARINGS OF THE COMMITTEE ON SCIENCE AND TECHNOLOGY, U.S. HOUSE OF REPRESENTATIVES, NINETY-FIFTH CONGRESS, SECOND SESSION, MARCH 21, 22, AND APRIL 4, 1978. Washington, DC; Committee on Science and Technology (1978). 442p. GPO.

Under Secretary of Energy Dale Myers said in the hearing on March 21, 1978, that the Administration's position is that Clinch River is not needed, but that technology should be kept in the forefront; that a program must be carried out that "means carrying on those pieces of Clinch River that apply to the technology from an engineering standpoint in terms of testing these various components. Further, have a design study which uses all of the top professional people in these various teams that have been associated with Clinch River, to aid in making design studies of those

things which are coming out of alternate fuel studies." At the March 22, 1978 hearing, Rep. Barry Goldwater, Jr. presented his views on the administration's proposal before proceeding with testimonies. He felt testimonials at the previous hearing did not include answers to "legal, financial, and other implications of terminating the Clinch River project." He said he was (1) concerned about the lengthy delay the abandonment of CRBR followed by a study of the new breeder would entail and (2) confused by the proposed coupling of fuel-cycle studies with plant design. He feels we should pay more attention to the plant itself, especially to such matters as performance, reliability, maintainability, and safety and worry about the fuel cycle later as a means of accelerating this project. At the April 4, 1978 hearing, panelists from industry, utilities, and university and public-interest personnel presented views on the future of the U.S. breeder program. (MCW)

- 4023 DESIGN OF THE FUELS AND MATERIALS EXAMINATION FACILITY AND MATERIALS HANDLING SYSTEMS. Doman, D.R. (Hanford Engineering Development Lab., Richland, WA); Trego, A.L.; Rogers, G.J.; Clark, H.E.; Peschel, W.E. Trans. Am. Nucl. Soc.; 30: 766-767(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
SPENT FUEL ELEMENTS; INSPECTION; REMOTE HANDLING EQUIPMENT; HOT CELLS; BREEDER REACTORS; FFTF REACTOR; DESIGN

- 4024 REMOTE EXAMINATION OF SHROUD TUBES IN LMFBR FUEL ELEMENTS. Ledbetter, J.M.; Serna, O.; Nicol, A.G.; Wood, W.T. (Los Alamos Scientific Lab., NM). Trans. Am. Nucl. Soc.; 30: 776-777(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
SPENT FUEL ELEMENTS; FUEL CANS; INSPECTION; REMOTE HANDLING EQUIPMENT; HOT CELLS; DESIGN; REMOTE HANDLING; LMFBR TYPE REACTORS

- 4025 EVALUATION OF HODOSCOPES FOR FUEL-MOTION MEASUREMENT IN MULTIPIN BUNDLES. Evans, A.E. Jr.; Orndoff, J.D.; Talbert, W.L. Jr. (Los Alamos Scientific Lab., NM). Trans. Am. Nucl. Soc.; 30: 465-466(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR CORE DISRUPTION; FAILED ELEMENT DETECTION; FUEL PINS; HODOSCOPES; PERFORMANCE; TREAT REACTOR

- 4026 IRRADIATION-SWELLING-INDUCED BOW IN FAST REACTOR SUBASSEMBLIES. Shields, J.A. Jr. (Argonne National Lab., Idaho Falls, ID). Trans. Am. Nucl. Soc.; 30: 154-155(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
LMFBR TYPE REACTORS; FUEL ASSEMBLIES; DEFORMATION; PHYSICAL RADIATION EFFECTS; FUEL CANS

- 4027 CONTROL OF OXYGEN, HYDROGEN, AND TRITIUM IN EBR-II SODIUM SYSTEMS. Osterhout, M.M. (Argonne National Lab., Idaho Falls, ID). Trans. Am. Nucl. Soc.; 30: 159-160(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).
EBR-2 REACTOR; PRIMARY COOLANT CIRCUITS; COLD TRAPS; PERFORMANCE; IMPURITIES; SECONDARY COOLANT

CIRCUITS; OXYGEN; HYDROGEN; TRITIUM

- 4028 STRAIN-RATE EFFECTS ON THE ELEVATED-TEMPERATURE BEHAVIOR OF ERNICK-3 WELD METAL. Klueh, R.L.; King, J.F. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 30: 157-158(1978). (CONF-7811109--).

From 1978 winter meeting of American Nuclear Society; Washington, DC, USA (12 Nov 1978).

CLINCH RIVER BREEDER REACTOR; STEAM GENERATORS; REACTOR MATERIALS; WELDED JOINTS; INCONEL 82; THERMAL STRESSES; YIELD STRENGTH; TENSILE PROPERTIES

- 4029 COMPUTER ANALYSIS AND CONTROL FOR NON-DESTRUCTIVE GAMMA SCANNING OF LMFBR FUEL PINS. Barnes, B.K.; Phillips, J.R.; Waterbury, G.R. (Los Alamos Scientific Lab., NM). Trans. Am. Nucl. Soc., Suppl.; 28: No. 1, 65-66(1978).

From Conference on computers in activation analysis and gamma-ray spectroscopy; Mayaguez, Puerto Rico (1 May 1978).

LMFBR TYPE REACTORS; FUEL PINS; GAMMA FUEL SCANNING; CAMAC SYSTEM; DATA PROCESSING

- 4030 FERRITIC STEELS FOR FAST REACTOR STEAM GENERATORS. Pugh, S.F.; Little, E.A. (eds.). London, England; British Nuclear Energy Society (1978). 247p. (CONF-770545--P1).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

Papers are presented on aspects of the technological development of ferritic steels for fast breeder reactors including physical and mechanical properties, behavior in sodium, corrosion by water, welding, and fabrication. (FS)

- 4031 FERRITIC STEELS FOR FAST REACTOR STEAM GENERATORS. Pugh, S.F.; Little, E.A. (eds.). London, England; British Nuclear Energy Society (1978). 272p. (CONF-770545--P2).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

Data are presented concerning ferritic steels for fast reactor steam generators. Separate abstracts were prepared for papers within the scope of the energy data base. (FS)

- 4032 ADVANCED LIQUID METAL FAST BREEDER REACTOR DESIGNS. Sayles, C.W. (Rockwell Int, At Int Div, Canoga Park, Calif). pp vp of Greater Los Angeles Area energy symposium. North Hollywood, CA; West Period Co. (1978).

From 4. energy symposium; Los Angeles, CA, USA (23 May 1978).

Fast Breeder reactor power plants in the 1000-1200 MW(e) range are being built overseas and are being designed in this country. While these reactors have many characteristics in common, a variety of different approaches have been adopted for some of the major features. Some of those alternatives are discussed.

- 4033 THERMAL EXPLOSION HAZARDS IN (FAST) NUCLEAR REACTORS. Zyszkowski, W. (Institute of Nuclear Research, Warsaw (Poland)). At. Energy Rev.; 16: No. 1, 3-87(1978).

The thermal explosion hazard is at the moment one of the crucial issues in the LMFBR safety programme that must be resolved before the commercial introduction of safe, reliable and economic fast breeder reactors. The present status of the understanding of thermal explosions and research activities is reviewed. The results of basic simulation experiments and

out-of-pile and in-pile engineering tests are discussed. The former give some insight into the physical phenomena, such as the triggering and propagation mechanisms of thermal explosion, and the latter reveal real peak pressures and the damage potential that can be expected during reactor accident conditions. The maximum loading of the reactor structure calculated with a variety of parametric models exceeds the containment capability of the present fast reactor design. On the other hand, a discrepancy exists between theory and the experimental evidence, and that, being the main point of the controversy, is discussed in the review.

- 4034 WATER CHEMISTRY IN THE DOUNREAY PROTOTYPE FAST REACTOR (PFR) POWER STATION. Melhuish, K.R.; Bray, J.A. (UKAEA, Dounreay. Nuclear Power Development Establishment). pp 63-76 of Water chemistry of nuclear reactor systems. London, England; Thomas Telford for BNES (1978).

From International conference on chemistry of nuclear reactor systems; Bournemouth, UK (24 Oct 1977).

The water treatment plants comprise three stages: a pre-treatment plant to deal with the high organic loading arising from the moorland catchment area; a make-up plant which was originally a twin stream demineralisation unit, each stream comprising cation bed, anion bed and mixed bed; and a condensate polishing plant of the 'deep bed' type with pre-coat filters. There are two 50% flow candle filter units followed by three 50% flow cation topped mixed bed units. The use of austenitic steels in the superheater and reheater requires close control from the drum water content for sodium, chloride and oxygen to minimise the risks of chloride or caustic stress corrosion.

- 4035 COMMERCIAL FAST REACTOR. Risley, UK; Nuclear Power Co. Ltd. (1978). 13p.

The design of the CFR is briefly described with sections on the primary circuit arrangement, fuel and reactor equipment handling, containment, steam generators, sodium pumps, intermediate heat exchangers and fuel and core layout.

- 4036 OPTIMISATION OF SODIUM-COOLED FAST REACTORS. London, England; British Nuclear Energy Society (1978). 464p. (CONF-771123--). British Nuclear Energy Society, London, England.

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

Separate abstracts were prepared for the individual papers presented at the meeting.

- 4037 SODIUM--NAK ENGINEERING HANDBOOK. VOLUME IV. SODIUM PUMPS, VALVES, PIPING, AND AUXILIARY EQUIPMENT. Foust, D.J. (ed.). Contract EY-76-C-03-0700. New York, NY; Gordon and Breach, Science Publishers, Inc. (1978). 296p. (SAN--0700-T1). Rockwell International Corp., Canoga Park, CA.

The handbook is useful for designers in the Liquid Metals Fast Breeder Reactor (LMFBR) program and by the engineering and scientific community performing investigation and experimentation requiring high-temperature Na and NaK technology. Data are presented for pumps, bearings and seals, valves, vessels and piping, and auxiliary equipment including vapor traps, freeze plugs, fuel-channel flow regulators, antivortexing devices, and miscellaneous mechanical elements. Reactor

materials are also discussed. (FS)

- 4038 FLUIDIZED BED HEAT TRANSPORT BETWEEN PARALLEL, HORIZONTAL TUBE-BUNDLES. (Westinghouse Electr Corp, Res and Dev Cent, Pittsburgh, Pa). pp 320-326 of Fluidization. Newby, R.A.; Keairns, D.L. New York, NY; Cambridge Univ. Press (1978).

From Conference on fluidization; Cambridge, UK (1 Apr 1978).

A fluidized bed heat transfer facility is described and results related to advanced fluidized bed heat exchanger concepts are discussed. Bed-to-tube heat transfer rates within compact tube-bundles and heat transport rates by bed mixing between parallel, horizontal tube-bundles are considered based on preliminary test results. The compact tube-bundles improve bubble distribution in the air-fluidized bed of 80 -M magnesia powder at velocities up to 30 cm/sec. Tube-to-bed heat transfer coefficients for centrally located tubes are generally increased by the presence of compact tube-bundles. A coefficient of bed transport is estimated for the tube-bundle configurations. This work is pertinent to electric power generation.

- 4039 SYSTEM MODELING FOR TRANSIENT ANALYSIS OF LOOP-TYPE LIQUID-METAL-COOLED FAST BREEDER REACTORS. Khatib-Rahbar, M. Ithaca, NY; Cornell Univ. (1978). 386p. University Microfilms Order No. 78-17,862.

Thesis (Ph. D.).

A digital computer simulation model for the dynamic analysis of multiloop liquid-metal-cooled fast breeder reactor plants is developed. The model is capable of simulating operational transients, anticipated incidents, postulated accidents which do not disturb the system geometry, and a major pipe rupture accident in the primary heat transport system. The mathematical model includes: point reactor kinetics, reactivity feedback, and reactor power control; primary, secondary, and tertiary thermal and coolant dynamics governed by forced and natural circulation; and plant protection and control systems.

- 4040 PROBABILISTIC METHODS IN REACTOR LICENSING. Tanguy, P. (Commissariat a l'Energie Atomique, Fontenay-aux-Roses, France). pp I.3.1-I.3.10 of Probabilistic analysis of nuclear reactor safety. La Grange Park, IL; American Nuclear Society, Inc. (1978).

From Probabilistic analysis of nuclear reactor safety; Los Angeles, CA, USA (8 May 1978).

Four years after the publishing of the WASH-1400 report, probabilistic methods still play little part in reactor licensing. One of the reasons could be the need for the general public of a maximum credible accident. Probabilistic methods will play a significant role in licensing only if they are introduced progressively and at different levels: assistance to safety assessments with present safety rules, introduction of new probabilistic safety criteria, definition of an overall risk acceptance level, use of operational experience feedback. The present situation in France is discussed, with the progress achieved and the difficulties met, both on PWR and on LMFBR. For the future, reactor licensing should be more probabilistic, but this requires an international consensus, which could be easily developed for the fast breeders.

- 4041 DEVELOPMENT OF A PROBABILISTIC MODEL OF

CONTAINMENT CONDITIONS FOLLOWING A CORE DISRUPTION ACCIDENT IN A BREEDER REACTOR. Peck, C.B. (General Electric Co., Sunnyvale, CA); Sultan, P. pp V.3.1-V.3.11 of Probabilistic analysis of nuclear reactor safety. La Grange Park, IL; American Nuclear Society, Inc. (1978).

From Probabilistic analysis of nuclear reactor safety; Los Angeles, CA, USA (8 May 1978).

Development of a probabilistic model for the evaluation of conditions in the containment of a breeder reactor following a core disruptive accident is discussed. The model as developed in the third phase of a risk assessment development program is presented. This model orders possible accident progression sequences, using various initial states, reaction modes and rates, and responses of engineered safety systems.

- 4042 RISK ASSESSMENT METHODS APPLICATION TO THE FAST FLUX TEST FACILITY. Feller, K. (General Electric Co., Sunnyvale, CA). pp V.4.1-V.4.12 of Probabilistic analysis of nuclear reactor safety. La Grange Park, IL; American Nuclear Society, Inc. (1978).

From Probabilistic analysis of nuclear reactor safety; Los Angeles, CA, USA (8 May 1978).

The technical approach and results of a risk assessment study based on the Fast Flux Test Facility reference design are summarized. The study was conducted as part of an ongoing effort to develop a technology base for breeder reactor risk assessment. The primary study objective was methods development. Due to the limited scope and lack of expert participation from many important technology areas, the results are not intended to represent the risk resulting from FFTF operation.

- 4043 APPLICATION OF JAMES-STEIN ESTIMATORS IN THE EVALUATION OF THE PHENIX REACTOR DATA BANK. Dubau, J.; Koen, B.V.; Georgin, J.P. pp VI.4.1-VI.4.8 of Probabilistic analysis of nuclear reactor safety. La Grange Park, IL; American Nuclear Society, Inc. (1978).

From Probabilistic analysis of nuclear reactor safety; Los Angeles, CA, USA (8 May 1978).

The James-Stein estimators are used to evaluate failure rate data for the French fast breeder reactor operating components. A comparison with other estimators is also performed.

- 4044 RISK ALLOCATION APPROACH TO REACTOR SAFETY DESIGN AND EVALUATION. Gokcek, O.; Temme, M.I.; Derby, S.L. (General Electric Co., Sunnyvale, CA). pp VII.3.1-VII.3.12 of Probabilistic analysis of nuclear reactor safety. La Grange Park, IL; American Nuclear Society, Inc. (1978).

From Probabilistic analysis of nuclear reactor safety; Los Angeles, CA, USA (8 May 1978).

This paper describes a risk allocation technique used for determining nuclear power plant design reliability requirements. The concept of risk allocation-optimum choice of safety function reliabilities under a maximum risk constraint - is described. An example of risk allocation is presented to demonstrate the application of the methodology.

- 4045 USE OF QUANTITATIVE RISK AND PROBABILISTIC SAFETY CRITERIA IN THE CONCEPTUAL DESIGN OF A LARGE POOL-TYPE LMFBR. Hartung, J.A.; Lancet, R.T. (Atomics International

Div., Canoga Park, CA). pp VII.4.1-VII.4.10 of Probabilistic analysis of nuclear reactor safety. La Grange Park, IL; American Nuclear Society, Inc. (1978).

From Probabilistic analysis of nuclear reactor safety; Los Angeles, CA, USA (8 May 1978).

This paper describes how quantitative risk and probabilistic safety criteria have been used to supplement licensing regulations to guide the safety-related design of a large (1000 MWe) pool-type LMFBR.

4046 DESIGN STAGE PROBABILISTIC ASSESSMENT OF PROTOTYPE LMFBR SAFETY SYSTEMS. Kondo, S.; Togo, Y. (Univ. of Tokyo, Japan). pp VII.6.1-VII.6.20 of Probabilistic analysis of nuclear reactor safety. La Grange Park, IL; American Nuclear Society, Inc. (1978).

From Probabilistic analysis of nuclear reactor safety; Los Angeles, CA, USA (8 May 1978).

Probabilistic assessment has been applied to the design of prototype liquid metal cooled fast breeder reactor safety systems. Natural convection cooling or sufficient length of allowable outage time has been found to be the most desirable feature for assuring high availability of the decay heat removal system. The allowable outage time has been evaluated from probabilistic point of view, and it has been found that large temperature margin is required to determine it due to the uncertainty in the decay heat curve and in the friction factor for fluid flow. For safe shutdown system, several designs of rod drive mechanism as well as structure of logic train have been compared from the reliability viewpoint. Highly reliable shutdown capability has been obtained by the combination of two independent systems. Some discussion has been devoted to the effect of grouping of instrument channels sets, cross-linking, and the reliability under the earthquake condition.

4047 ROLE OF DISTRIBUTED MICROSCOPIC PHYSICAL PHENOMENA IN PROBABILISTIC ACCIDENT ANALYSIS WITH DETERMINISTIC CODES. Abramson, P.B.; Hummel, H.H.; Gelbard, E.M.; Pizzica, P.A.; Sienicki, J.J. (Argonne National Lab., IL). pp VIII.4.1-VIII.4.13 of Probabilistic analysis of nuclear reactor safety. La Grange Park, IL; American Nuclear Society, Inc. (1978).

From Probabilistic analysis of nuclear reactor safety; Los Angeles, CA, USA (8 May 1978).

Analysis of Hypothetical Core Disruptive Accidents with large computer codes is based upon the assumption that enough basic physical information is available to enable the analyst to write down the proper analytical description of these processes and that macroscopic behavior will be properly calculated as an end product. The thermodynamic and hydrodynamic processes that go on during such a disrupted state analysis are dependent upon the geometry and flow regime, which are unknown and not deterministically predictable with any ease. Thus analysts have associated variable parameters with these processes and the sensitivity of macroscopic results to the microscopic parameters has been examined by studying the response function of a straight through calculation to a different parametric value. However, the processes depend upon functions which are distributed rather than single valued. Analysis which adds in the information about the distributed nature of the microscopic processes reduces the sensitivity of the macroscopic result to the microscopic parameter.

4048 GRAVITATIONAL AGGLOMERATION OF POST-HCDA LMFBR AEROSOLS. Partner, G.A. Columbia, MD; Univ. of Missouri (1978). 256p. University Microfilms Order No. 7906908.

Thesis (Ph. D.).

A Hypothetical Core Disruptive Accident (HCDA) in an LMFBR may release a very complex and radioactive aerosol into the primary containment atmosphere. After release, the aerosol particles will settle and there will be particle-particle collisions. Coagulation between the particles leads to the formation of a third, larger aerosol particle. The coagulation process of aerosols is stipulated to be composed of three separation mechanisms - Brownian, turbulent, and gravitational coagulation. This dissertation is concerned only with gravitational coagulation, which occurs due to the differential settling velocities of two aerosol particles. Results are presented for the calculation of the gravitational collision efficiency for various conditions. Both constant density and size dependent density aerosol particles are analyzed. The larger particle radius is varied from 0.5 μ to 100 μ , and appropriate drag force terms are utilized. Also, since the containment atmospheric properties may vary during the course of an accident, the effect on epsilon of changing the fluid properties is investigated. The results show that both the particle size and density have a strong effect on the gravitational collision efficiency. In general, it is difficult to construct a simple representation of epsilon as a function of these variables.

4049 BUOYANCY EFFECTS ON SODIUM COOLANT TEMPERATURE PROFILES MEASURED IN AN ELECTRICALLY HEATED MOCK-UP OF A 61-ROD BREEDER REACTOR BLANKET ASSEMBLY. Engel, F.C.; Markley, R.A.; Minushkin, B. (Westinghouse Electr. Corp., Madison, Pa). Am. Soc. Mech. Eng., [Pap.]; No. 78-WA/HT-25, 1-14(1978).

This paper describes test results demonstrating the effect of buoyancy on the temperature profiles in a 61-rod electrically heated mock-up of an LMFBR radial blanket assembly. In these assemblies, heat transfer occurs over a wide range of complex operating conditions. The range and complexity of conditions are the result of the steep flux and power gradients which are an inherent feature of the blanket region and of the power generation level in an assembly which can vary from 20 to 1100 kw.

4050 SIMPLE FUEL PIN TRANSIENT AND MELTING MODEL AND ITS APPLICATION TO THERMAL-HYDRAULICS IN LMFBR SUBASSEMBLY. Chen, W.L.; Ishii, M.; Groimes, M.A. (Argonne Natl. Lab., Ill). Am. Soc. Mech. Eng., [Pap.]; No. 78-WA/HT-26, 1-16(1978).

A simple conduction model with phase change has been developed for the transient analysis of a reactor fuel pin based on lumped parameter techniques. The purpose of this analysis is to provide a simple useful tool to obtain the general information about fuel and clad leading into the cooling transients and melting. Such a simple fuel and clad thermal transient model is particularly useful to multichannel analysis where conventional conduction computer codes require considerable computing time and storage space. At present, this formulation is being employed for the analysis of sodium thermohydraulics, sodium voiding, and melting of cladding and fuel in a subassembly of a fast reactor core. A detailed analysis of predicted coolant, fuel and cladding thermal transients leading into sodium voiding and fuel pin

melting has been made in comparison with the results of various in-pile experiments and with the predictions from the existing more complicated codes.

- 4051 PAST AND PRESENT ROLE OF FAST BREEDER REACTORS IN JAPAN. Tomabechi, K. (Power Reactor and Nuclear Fuel Development Corp., Tokyo, Japan). pp 15-27 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

To meet increasing future energy demand in Japan utilization of fission energy should be promoted. In particular it is of vital importance to develop and utilize FBRs as soon as possible in order to save the natural uranium needed. Ten years have passed since the development of FBRs in Japan was initiated as a national project with the highest priority and now the experimental fast reactor JOYO is successfully being commissioned and the prototype fast breeder reactor MONJU has reached the stage of proceeding to construction. Experience gained from past development work encourages further strong effort towards the commercialization of FBRs. The important role to be played by FBRs, the development programme for FBRs, and the experience gained so far from the programme in Japan are reviewed.

- 4052 PAST AND PRESENT ROLE OF FAST BREEDER REACTORS IN THE FEDERAL REPUBLIC OF GERMANY. Guthmann, E. (INTERATOM, Bergisch Gladbach, Germany); Kessler, G.; Stoehr, H. pp 57-68 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

Studies on future energy needs, the introduction of light water reactors and the availability of uranium on the world market lead to the conclusion that the fast breeder should be available for commercial introduction after 1990. The fast breeder development programme in the FRG, Belgium, and The Netherlands is concentrating on these project aims. At present the KNK II test reactor is in operation and the 300 MW(e) prototype fast reactor SNR 300 is under construction. It is intended to build a future commercial-size fast breeder in close co-operation with Belgium, The Netherlands, France, and Italy.

- 4053 PAST AND PRESENT ROLE OF FAST BREEDER REACTORS IN FRANCE: EXPERIENCE GAINED WITH RAPISODIE AND PHENIX. Conte, F. (CEA, Centre de Marcoule, Bagnols-sur-Cèze, France); Brosseau, G.; Pontier, R. pp 69-78 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In French)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

Rapisodie has recently celebrated ten years of operation, which provides an occasion for reviewing the performance of the facility--availability, behaviour of the main components, experiments carried out, evolution of operating conditions, and difficulties encountered. A

major operation recently carried out successfully, i.e. the replacement of the control plug, is described. The experience gained during four years of industrial operation of the Phenix facility is also discussed. An account is given of the behaviour of the material, fuel performance (burn-up, number of assemblies irradiated, breeding ratio), the dosimetry and the wastes produced. Among the difficulties encountered the main event has been a sodium leak on the intermediate heat exchangers; all six exchangers have been repaired under satisfactory working conditions, without causing any irradiation of the personnel.

- 4054 PAST AND PRESENT ROLE OF FAST BREEDER REACTORS IN THE USSR. Kochetkov, L.A.; Bagdasarov, Yu.E. (Inst. of Physics and Power Engineering, Obninsk, USSR). pp 79-87 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In Russian)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The development of sodium-cooled fast reactors in the USSR is reviewed. First the results of general research, and of the development and operation of experimental facilities, test rigs and reactors are briefly discussed--work which made it possible to formulate the basic principles for the design and construction of demonstration power plants such as BN-350. The experience derived from the construction and operation of BN-350 and from the construction of BN-600 is discussed. The experience of start-up, adjustment, and operation of BN-350 shows that sodium-cooled fast reactors have successfully completed the stage of industrial-scale pilot operation. The characteristics of the planned series of BN-1600 power plants are mentioned and compared with those of BN-350 and BN-600.

- 4055 PAST AND PRESENT ROLE OF FAST BREEDER REACTORS IN THE UNITED KINGDOM. Moore, J. (United Kingdom Atomic Energy Authority, Risley, England). pp 89-103 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

Predictions of requirements for energy and availability of primary energy sources in the UK during the next few decades show a strong incentive for early introduction of fast reactors. After the turn of the century they appear to be the only known method for meeting long-term ore that would be necessary for an on-going thermal reactor programme. Current proposals are to construct a commercial demonstration fast reactor, CDFR, with its fuel plants, to obtain the experience and confidence that would be necessary to launch a programme of fast reactors. Confidence in ability to proceed with CDFR is based on a substantial programme of development during the past 25 years culminating in the prototype fast reactor, PFR, and its fuel fabrication and reprocessing plants. Experience with PFR has been invaluable in guiding the design of CDFR.

- 4056 PAST AND PRESENT ROLE OF FAST BREEDER REACTORS IN ITALY. Castelli, G. (ENEL, Rome, Italy); Cicognani, G.; Ghilardotti, G.; Musso,

B. pp 105-114 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The programme that is under development in Italy for fast breeder reactors is described. The Italian engagement in the construction of the Creys-Malville plant is discussed as well as the work at present under way in R and D activities and on the PEC reactor. The roles of the different organizations involved in the fast breeder reactor programme are also considered with particular attention to the activities carried out by AGIP NUCLEAR, CNEN, ENEL, NIRA, and by the manufacturing companies. An overall picture of the different agreements between European countries is also given with reference to the construction of the Creys-Malville plant and to research and industrial development activities.

- 4057 PAST AND PRESENT ROLE OF FAST BREEDER REACTORS. Megy, J. (Centre d'Etudes Nucleaires de Saclay, Gif-sur-Yvette, France). pp 115-139 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In French)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The LMFBR type reactor development programs of the UK, Japan, USSR, USA, German Federal Republic, France, and Italy are reviewed.

- 4058 MEASUREMENTS IN PFR OF IMPORTANCE TO SAFETY AND REACTOR PERFORMANCE. Smith, D.C.G. (Dounreay Nuclear Power Development Establishment, Caithness, England); Crowe, D.S.; Lennox, T.A.; Hampshire, R. pp 143-158 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

Measured values of a number of important parameters associated with PFR core performance are given and compared with calculations. Generally agreement is good but where discrepancies have been found attention is drawn to possible reasons.

- 4059 SHUTDOWN SYSTEMS OF KNK AND SNR 300. Glinsky, H.W.; Mehren, F.; Voj, P.; Hoffmann, H.; Marten, K. (INTERATOM, Bergisch Gladbach, Germany). pp 191-203 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The shutdown systems are an integral part of the reactor protection system. Two independent systems are installed to increase the reactor safety. These systems consist of several separate units which operate in different ways to avoid common mode failures. One of the shutdown systems also controls the reactor. The development of the shutdown systems is supported by an extensive experimental programme: tests with components; tests with prototypes in water and sodium; tests with

units which are to be installed in the reactor; and a final operational test in the reactor. The principle of the KNK II shutdown systems is described.

- 4060 OPERATING EXPERIENCE OF THE BN-350 FAST REACTOR (1972-1977). Yurchenko, D.S.; Vasilenko, K.T.; Demin, A.A.; Pomerantsev, G.B.; Samarkin, A.A.; Skorikov, N.V.; Tverdokskii, N.D.; Shkol'nik, V.S. (BN-350 Reactor, Shevchenko, USSR). pp 227-246 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In Russian)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The start-up and power commissioning of the BN-350 reactor are reported together with the results of experimental studies. The main results of operation from the time the reactor reached a power level of 650 MW(th) till 1 January 1978 are given. Operating experience with the principal equipment and technological system is discussed and radiation conditions are described. The cases of failure of the steam generator heat-transfer surfaces are considered.

- 4061 THEORETICAL AND EXPERIMENTAL STUDIES ON THE PHYSICAL CHARACTERISTICS OF THE BN-350 REACTOR. Alekhin, L.A.; Zvonarev, A.V.; Ivanov, A.P. (Inst. of Physics and Power Engineering, Obninsk, USSR). pp 263-285 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In Russian)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The main results of physics studies on the BN-350 reactor which were carried out during physical and power commissioning and in the early stages of operation are discussed. The characteristics considered were the critical parameters, the efficiency of the power control and instrumentation system components, the temperature and power coefficients of reactivity, reactivity effects due to fuel burn-up, the distribution of various reaction rates over the reactor, and the ratios of the average cross-sections of the fissionable isotopes. The calculated results were obtained by different methods and computation programs. It is shown that the theoretical approaches developed during the design of the reactor, together with the BNAB-70 multigroup constants, ensure good agreement between the calculated and experimental values in most cases.

- 4062 PRIMARY CIRCUITS OF PHENIX AND SUPER-PHENIX. Vial, B. (Centre de Marcoule, Bagnols-sur-Geze, France); Thevenin, M.; Guidez, J.; Sauvage, M.; Schaler, K. pp 295-312 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In French)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

An account of the operating experience of the primary circuits of Phenix since it went into industrial-scale operation is presented. After a brief description of the primary circuits, the principal incidents which have

interfered with reactor operation, mainly those which affected the pumps and intermediate exchangers, are mentioned. Detailed comments are made of how the operations of removal, repair and re-installation of the intermediate exchangers were carried out. The second part gives a comparative description of the Phenix and Super-Phenix primary circuits. The reasons which led to design changes in Super-Phenix are explained. The most important modifications involved the reactor cover and the type of heat insulation used, the supporting structure of the plate resting on the bottom of the main vessel and the use of double recess.

- 4063 COMPARATIVE ANALYSIS OF LAYOUT AND DESIGN IN THE RN-600 AND BN-350 REACTORS. Makovskii, A.A.; Mitenkov, F.M.; Pakhomov, V.V.; Shiryayev, V.I. (Gorkii Polytechnic Inst., USSR). pp 313-325 of Design, construction, and operating experience demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In Russian)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The technical solutions adopted for the primary circuit, individual items of equipment and the design of the most important components of the BN-350 and BN-600 reactors emphasizing layout characteristics are presented. The advantages and disadvantages of the loop and pool designs are discussed with regard to the reliability and safety of these reactors, and the experience acquired in building them. The technical advantages of the pool design are substantiated for fast reactors in the medium power range.

- 4064 PROBLEMS OF DESIGNING AND TESTING PRIMARY LOOP COMPONENTS OF FAT REACTORS. Braggamoli, F. (CEN, Bologna, Italy); Tonarelli, L.; Bandiera, P.G. pp 327-335 of Design, construction, and operating experience demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In Italian)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The main problems encountered in Italy in connection with designing and testing the primary loop components of the PEC, the Creys-Malville power plant, and the Super-Phenix line are described. The work carried out on piping, intermediate heat exchangers, drives, mechanical pumps, and inflatable joints, and also the present status of developmental work on sodium test loops for this type of component are described.

- 4065 DEVELOPMENT, DESIGN, CONSTRUCTION, AND TESTING OF PUMPS, INTERMEDIATE HEAT EXCHANGERS AND MANIPULATOR SYSTEMS FOR IN-SERVICE INSPECTION OF SNR 300. Vroom, J.P. (B.V. Neratoom, The Hague, Netherlands); Heslenfeld, M.W.; Westerweele, W.J.; Hoeft, J.E.; Kirchner, G.F.; Menck, J.W.; van Willigen, A. pp 347-374 of Design, construction, and operating experience demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The development, design, construction, and testing activities for sodium pumps and intermediate heat exchangers as carried out by

NERATOOM in the Netherlands are described. Testing of full-size components under operational conditions yielded important information for the design of the components for SNR 300. Information is also given in brief on the manufacture of the components. The second part reviews the work done on manipulator systems for in-service inspection as carried out by INTERATOM in Germany. In this field again experience gained from tests had important consequences for the design of the apparatus for SNR 300.

- 4066 DESIGN AND EXPERIMENTAL TESTING OF INTERMEDIATE HEAT EXCHANGERS. Budov, V.M.; Golovko, V.F.; Grddichko, E.M.; Lukasevich, B.I.; Loginov, S.A.; Sabin, M.I. pp 375-392 of Design, construction, and operating experience demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In Russian)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The design and principal characteristics of the intermediate heat exchangers selected for the BN-350 and BN-600 reactors are presented. The problems which arose during the design of the heat exchangers and the approach adopted for their solution are considered. The experimental testing of heat exchangers, involving the study of thermal, hydrodynamic, and vibrational characteristics, is discussed.

- 4067 OPERATING EXPERIENCE WITH THE PFR PRIMARY CIRCUIT AND DESIGN INTENTIONS FOR CDFR. Broadley, D. (Nuclear Power Co. (Risley) Ltd., England); Laithwaite, J.M.; Walford, J.G. pp 403-410 of Design, construction, and operating experience demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The high reliability of the reactor and the primary circuit of PFR is noted. Despite a small number of difficulties associated with the auxiliaries the operation of the reactor has been unaffected. The current status of the CDFR design is described. The influence on this design by the substantial and valuable experience gained from PFR is discussed.

- 4068 FUEL AND COMPONENT HANDLING IN PFR AND INTENTIONS FOR CDFR. Broomfield, A.M. (Dounreay Nuclear Power Development Establishment, Caithness, England); Holmes, J.A.G.; Webb, J. pp 425-433 of Design, construction, and operating experience demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The extensive operating experience with fuel and component handling in the Prototype Fast Reactor (PFR) is summarized, with emphasis on the fuel transfer system. The route for fuel sub-assemblies, including preparation in the new fuel cell, transfer to the storage rotor within the reactor vessel, transfer between rotor and core and the return to the irradiated fuel cave is described with details of experience so far. The cave was commissioned in 1977 and has received the first irradiated fuel from the reactor for examination. A large number of sub-assemblies has been handled

between storage rotor and core without significant difficulty and the design rates of transfer have been achieved. The fuel charge machine must be loaded into the reactor for each refuelling operation. The design of a machine which can remain in the reactor during operation is being prepared, since experience indicates that the time for refuelling shutdowns can be substantially reduced.

- 4069 ELECTROCHEMICAL AND PHYSICAL METHODS FOR REACTOR DIAGNOSIS. Fuge, R.; Teske, K.; Ullmann, H. (Central Inst. for Nuclear Research, Rossendorf, Germany). pp 435-449 of Design, construction, and operating experience demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

Electrochemical cells with solid electrolyte have been developed to measure the oxygen and the hydrogen concentration in sodium and cover gas. Experimental results are presented and the method is recommended for the detection of water leaks in steam generators. A supplementary method for the leak detection is the "bubble noise" in magnetic flowmeter output signals. Hydrogen and argon void fractions of one per cent can be detected. Some results of R and D work for the acoustic monitoring of the reactor and the steam generator are presented.

- 4070 FUEL HANDLING EXPERIENCE WITH LIQUID-METAL REACTORS. Foster, K.W. (Atomics International, Canoga Park, CA). pp 451-464 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The basic design approach for fuel-handling equipment for Liquid Metal Fast Breeder Reactors in the USA originated from the Experimental Breeder Reactors (EBR) I and II and the Sodium Reactor Experiment (SRE). EBR-II paved the way for many of the current in-vessel handling concepts, such as rotating plugs, and although the SRE was a thermal reactor, the mechanical design of ex-vessel handling machines evolved from this concept. Early fuel-handling system design concepts are described. Design, development, fabrication, and operating problems that led to the next design generation are discussed. Reactor concepts discussed include SRE, HNPFF, EBR-II, Fermi, SEFOR, FFTF, and CRBRP. Also, concepts under consideration for the commercial size Prototype Large Breeder Reactors (PLBR) are covered.

- 4071 TANK FOR STORAGE OF SPENT FUEL ELEMENTS UNDER SODIUM: SPECIAL DESIGN FEATURES AND SOLUTIONS PROPOSED. Debauche, M.; Dehon, C.; Brahy, N. (Belgonucleaire, Brussels, Belgium). pp 477-488 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In French)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

A short explanation is given of the handling circuit for the SNR fuel. In the description of the storage facility the emphasis is on the components, which are examined in detail. A

brief reference to their functions is followed by a description of the solution adopted and a discussion of its advantage.

- 4072 CONTRIBUTION OF ITALIAN INDUSTRY TO THE DEVELOPMENT OF REACTOR COMPONENTS AND FUEL-HANDLING SYSTEMS OF FAST REACTORS. Giordano, P. (NIRA, SpA, Genoa, Italy); Di Leo, S.; Berni Canani, S.; Piana, F. pp 489-505 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The principal activities carried on by Italian industry concerning in-core structures, core facilities, and fuel-handling systems are described. In particular the following items are described: the PEC reactor central experimental zone equipment (test channel), whose design is in progress at NIRA; the core support structures for Super-Phenix 1, which will be constructed by BREDA; the out-of-core fuel-handling system of PEC, delegated to SNIA-TECHINT, who have now completed the feasibility studies and are proceeding to construction plans; the parts of the primary fuel-handling system for Super-Phenix 1, i.e. charging and discharging machines and the straight-pull machine, designed and made by NIRA and CMI.

- 4073 OPERATION OF THE PFR AND DESIGN OF THE CDFR SECONDARY CIRCUIT. Taylor, D. (Nuclear Power Co. (Risley) Ltd., England); Smedley, J.; Broomfield, A.M. pp 509-525 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The design and operational experience of the Prototype Fast Reactor (PFR) secondary circuits are reviewed. The preparation of the design proposal for this development strategy is the intention to install Replacement Tube Bundles (RTB) in the superheater and reheater shells, which will be virtually full-size prototypes of the CDFR superheater units.

- 4074 DESIGN, CONSTRUCTION, AND OPERATING EXPERIENCE OF SECONDARY SODIUM CIRCUITS OF FAST REACTORS. Robin, M.G. (Centre d'Etudes Nucleaires de Saclay, Gif-sur-Yvette, France); Saur, J.M.; Vial, B.; Brachet, A. pp 527-555 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In French)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The designs of the secondary sodium circuits of fast reactors developed in France are presented. Each sodium loop comprises the following main components: two intermediate heat exchangers in which the secondary sodium circulates inside the tubes; a steam generator (single passage of water) which may be either of the multi-module (Phenix) or of the single-module (Super-Phenix) type; and a vertical mechanical pump immersed in the sodium expansion tank. Main characteristics, date of sodium filling and duration of power operation are specified for the Phenix. A more detailed description is given of the incidents which have occurred and of the measures taken to

prevent their recurrence; also of the main modifications to the circuits. The means of monitoring the physico-chemical characteristics of sodium are listed and the principal findings as regards its purity are presented.

- 4075 DESIGN OF THE MONJU HEAT TRANSPORT SYSTEM AND THE RELATED R AND D WORK, INCLUDING STEAM GENERATOR. Kanamori, A.; Imanaka, N.; Nakai, Y. (Power Reactor and Nuclear Fuel Development Corp., Tokyo, Japan). pp 573-585 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The heat transport system of the prototype plant MONJU is based on the 'loop-type' concept and consists of three circuits of both primary and intermediate loops. The main pumps are located in the cold leg in both loops and the intermediate heat exchangers are of the vertical shell and tube type with no sodium level. The steam generators are composed of helically coiled tubes and are divided, with superheater and evaporator, both of which contain cover gas. Various kinds of R and D work have been conducted since 1970 to demonstrate the feasibility of the components and the experimental results have been reflected in detailed designs.

- 4076 DEVELOPMENT, DESIGN, CONSTRUCTION, AND PROTOTYPE TEST EXPERIENCE OF STEAM GENERATORS FOR SNR 300. Knaap, M.H. (B.V. NERATOOM, The Hague, Netherlands); Ludwig, P.W.P.H.; Smit, C.C.; Van Westenbrugge, M.K.; Broodman, J.J.; Vander Krogt, C.A.J. pp 587-603 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

Development and construction of the steam generators for the SNR 300 LMFBR are described. Basic design, test results of the prototypes and experiences with the licensing process are described. The test of the straight-tube type made some design changes necessary; the test of the helical-tube type was flawless. During testing of both the straight-tube and the helical-tube steam generator spurious hydrogen signals occurred.

- 4077 DESIGN, R AND D, AND MANUFACTURE OF THE COMPONENTS OF SECONDARY CIRCUITS FOR FAST SODIUM POWER PLANT. Avanzini, P.G. (NIRA, Genoa, Italy); Valentini, P.; Bottiglioni, F. pp 605-627 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The efforts of Italian industry in the field of components for fast sodium power plants are presented. The main components involved are steam generators, sodium/air heat exchangers and large sodium valves. The development work carried out for these components is outlined and the main realizations are presented. A brief account is also given of the current situation of sodium plants supporting the development of these components. Research on

materials and the development of special sodium components are also reported upon.

- 4078 PROBLEMS OF SODIUM/WATER STEAM GENERATOR SAFETY AND THEIR SOLUTION IN THE USSR. Poplavskii, V.M.; Bagdasarov, Yu.E.; Kozlov, F.A.; Kochetkov, L.A.; Titov, V.F. (Inst. of Physics and Power Engineering, Obninsk, USSR). pp 629-644 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In Russian)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The main results of experimental and theoretical studies on emergency conditions associated with sodium/water interaction are discussed. The current Soviet approach to the safety problems of sodium/water steam generators with single-wall separation of coolants is described.

- 4079 EXPERIMENTS ON THE FLUID DYNAMICS AND THERMODYNAMICS OF ROD BUNDLES TO VERIFY AND SUPPORT THE DESIGN OF SNR-300 FUEL ELEMENTS: STATUS AND OPEN PROBLEMS. Moeller, R.; Weinberg, D.; Trippi, G.; Tschoeke, H. (Kernforschungszentrum, Karlsruhe, Germany). pp 673-692 of Design, construction, and operating of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The reliable design of reactor core elements calls for precise knowledge of the 3D-temperature fields of the different components; this primarily applies to the fuel element cladding tubes, these being the first safety barrier. Where the 3D-temperature fields so far determined exclusively with the help of global thermohydraulic computer codes (SUBCHANNEL-Codes) have to be determined more accurately by local investigations is analyzed. The basis of these investigations is the measurement of local velocities and temperature in 19-rod bundle models of the SNR-300 fuel element performed at the Kernforschungszentrum Karlsruhe (KfK). Some important results of the extensive experimental investigations are reported and compared with global and local recalculations. Open problems are pointed out. The influence of the uncertainties in the thermohydraulic design with respect to the strength analysis are discussed.

- 4080 DESIGN, CONSTRUCTION, AND OPERATING EXPERIENCE OF DEMONSTRATION LMFBR. THE APPLICATION OF CORE AND FUEL PERFORMANCE EXPERIENCE IN BRITISH REACTORS TO COMMERCIAL FAST REACTOR DESIGN. Bagley, K.Q. (UKAEA, Risley, England); Gatley, J.A. pp 725-736 of Design, construction, and operating of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The Prototype Fast Reactor (PFR) sub-assembly design is described with particular emphasis on the choice of factors that are important in determining satisfactory performance. Reasons for the adoption of specific clad and fuel design details are given in their historical context, and irradiation experience--mostly from the Dounreay Fast

Reactor (DFR)--in support of choices is described. The advantages of restrained core and the application of this concept to the Commercial Demonstration Fast Reactor (CDFR) core design is described. Probable future trends in pin and sub-assembly design are reviewed and the scope of associated irradiation testing programmes defined. Arrangements for monitoring and evaluating fuel performance, both in reactor and post-irradiation, are outlined and the provisions for endorsement of CDFR pin, sub-assembly and core design details in PFR are indicated.

- 4081 FUTURE PLANS FOR THE DESIGN AND CONSTRUCTION OF FAST REACTOR POWER STATIONS. Moore, J. (Risley Nuclear Power Development Establishment, England). pp 739-751 of Design, construction, and operating of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The development programs for LMFBR type reactors in France, Germany Federal Republic, Italy, Japan, The UK, the USA, and USSR are reviewed.

- 4082 FUTURE PLANS FOR THE DESIGN AND CONSTRUCTION OF FAST BREEDER REACTORS IN FRANCE. Benmergui, A. (Electricite de France, Paris-La-Defense); Megy, J.; Vautre, L.; Villeneuve, J. pp 753-773 of Design, construction, and operating of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In French)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The motives underlying the development of fast neutron reactors in France and the chief characteristics of Super-Phenix together with the state of construction of the power station are discussed. Prospects for the construction of fast breeder reactors in France following Super-Phenix are analyzed. There is reference to the industrial organization and the European agreements established for the purpose. A detailed description of Super-Phenix (Creys-Malville Power Station) is given.

- 4083 FUTURE PLANS FOR THE DESIGN AND CONSTRUCTION OF FAST REACTOR POWER STATIONS IN THE FEDERAL REPUBLIC OF GERMANY. Kempken, M. (Federal Ministry for Research and Technology, Bonn, Germany); Koehler, M.; Wolfe, M. pp 775-783 of Design, construction, and operating of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

Some important design features of future fast reactors in the Federal Republic of Germany (FRG) are presented, in particular for the demonstration plant SNR 2 which is to follow the prototype SNR 300, presently under construction in Kalkar. The SNR 2 conceptual design will be based on the SNR 300 design as far as possible. Programmes for the introduction of fast breeder reactor power stations on the part of the governments, the utilities, and suppliers are based on broad international co-operation.

- 4084 FUTURE PLANS FOR THE DESIGN AND

CONSTRUCTION OF FAST REACTOR POWER STATIONS IN ITALY. Castelli, G. (ENEL, Rome, Italy); Ghilardotti, G.; Musso, B.; Pierantoni, F. pp 785-792 of Design, construction, and operating of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

Studies related to fast reactor technology have been pursued in Italy for a long time and this country is now deeply engaged in the demonstration and marketing phases, in accordance with the outlines of the Italian national energy plan. The following topics are examined: current possibilities for introducing fast reactors in Italy; the main social and political constraints concerning their introduction; the necessary industrial and organizational structures (in the broadest meaning) existing or foreseen; the national programme pertaining to activities towards achieving this goal.

- 4085 FUTURE PLANS FOR THE DESIGN AND CONSTRUCTION OF FAST REACTOR POWER STATIONS IN JAPAN. Tomabechi, K.; Miki, R. (Power Reactor and Nuclear Fuel Development Corp., Tokyo, Japan). pp 793-804 of Design, construction, and operating of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

Tests for increasing the power of the experimental fast reactor JOYO which went critical in 1977 will begin soon and the maximum power of 50 MW(t) will be attained in the middle of 1978. Construction of the prototype reactor MONJU of 300 MW(e) will begin in 1979 with a schedule of operation in the mid-1980s. Following operation on MONJU, construction of a large demonstration reactor of 1000 to 1500 MW(e) will be undertaken. Some 2 to 3 years after the construction of the demonstration reactor, a series of reactors will be constructed similar in size and design to promote commercialization of LMFBRs. The future programme of developing LMFBRs is considered the most desirable way to promote commercialization of LMFBRs and strong efforts will be made to put this programme into practice. It is expected that LMFBRs will play an important role in mitigating the serious problem of energy supply in Japan foreseeable around the turn of the century.

- 4086 FUTURE PLANS FOR THE DESIGN AND CONSTRUCTION OF FAST REACTOR POWER STATIONS IN THE UNITED KINGDOM. Vaughan, R.D.; Wolff, P.H.W. (Nuclear Power Co. (Risley) Ltd., England). pp 805-816 of Design, construction, and operating of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978).

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

There is to be held in the United Kingdom a public inquiry into the need for fast reactors, and specifically the construction of a demonstration plant which must precede a commercial breeder programme. In anticipation of a successful outcome of this inquiry, current attention is directed toward the objectives, design, and construction of the demonstration reactor. The technical objectives of this reactor are: to validate all technical

features of the first commercial reactors; to optimize operating techniques; to provide long-term information in advance of the commercial projects, and in due course to provide a vehicle for testing developments needed for later breeder programmes. Construction of the reactor will also establish in the UK the technical and manufacturing competence, the project organization and financial relationships necessary for a successful commercial programme.

- 4087 FUTURE PLANS FOR THE DESIGN AND CONSTRUCTION OF FAST REACTOR POWER STATIONS IN THE USSR. Kazachkovskii, O.D.; Mitenkov, F.M.; Troyanov, M.F. (Inst. of Physics and Power Engineering, Obninsk, USSR). pp 817-827 of Design, construction, and operating of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In Russian)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The principal structural, heat-engineering and physical characteristics of a fast reactor intended for a 1600 MW(e) power unit are presented. Sodium is used as coolant in the primary and secondary circuits and water and steam in the tertiary circuit. The reactor is so designed as to ensure high fuel cycle performance and economic power generation.

- 4088 REPLACEMENT OF THE RAPSODIE CONTROL PLUG: EXPERIENCE WITH AN IMPORTANT OPERATION. Delisle, J.P.; Allegre, R.; Besoux, M.; Laland, R.; Ollivier, J.; Snrech, D. (Centre d'Etudes Nucleaires de Cadarache, France). pp 413-423 of Design, construction, and operating experience demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In French)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The Rapsodie control plug is a 4.5 tonne structure located above the core at the center of the reactor and thus highly radioactive. It was replaced in 1977 after several years of study. The operation, which was carried out without much difficulty, and the conclusions that could be drawn are described.

- 4089 STUDY OF THE OPERATIONAL CHARACTERISTICS OF THE CORE AND IRRADIATED FUEL PARAMETERS IN THE BOR-60 REACTOR. Gryazev, V.M.; Afanas'ev, V.A.; Gadzhiev, G.I.; Efimov, V.N.; Inchagov, A.V.; Narkin, S.A.; Nigmatullin, N.R.; Teikovtsev, A.A. (All-Union Research Inst. of Nuclear Reactors, Dimitrovgrad, USSR). pp 659-672 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In Russian)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The results are presented of studies on the physical, thermophysical, and technological characteristics of the core and on the parameters of irradiated fuel of the BOR-60 reactor for the 1970-1977 operational period. The main purpose of BOR-60 operation was to gain experience of reliable functioning of the basic equipment and to carry out large-scale tests on fuel elements and structural materials to be used in demonstration fast reactors. BOR-60 has been used for an extensive programme of

irradiation of standard fuel elements based on 90% enriched uranium dioxide. Studies are being conducted on assemblies with vibro-compacted fuel, mixed oxide fuel, cladding steels of different grades on so on--all of which are necessary if the efficiency of the fuel cycle is to be improved in future fast reactors.

- 4090 SAFETY ASPECTS OF FAST REACTOR CORE OPERATION. Gourdon, J. (Centre d'Etudes Nucleaires de Cadarache, Saint-Paul-lez-Durance, France); Penet, F.; Savineau, M. pp 159-175 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In French)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

On the basis of the operating experience for Phenix, procedures are described for monitoring the main safety parameters of the core without appreciably disturbing the power plant operation. Guarding against errors of core loading during handling operations is discussed. The methods and the results of checking control rod efficiency and following up the core reactivity and the neutron negative feedback factors are presented. The preventive measures, spread over the handling chain, which were effective in preventing any core loading error during three years of operation are indicated.

- 4091 SAFE OPERATION OF THE BOR-60 EXPERIMENTAL FAST REACTOR IN POWER PLANT CONDITIONS. Gryazev, V.M.; Antipin, G.K.; Afanas'ev, V.A.; Britov, V.N.; Efimov, V.N.; Kondrat'ev, V.I.; Kon'kov, V.F.; Smirnov, A.M. (All-Union Scientific Research Inst. of Nuclear Reactors, Dimitrovgrad, USSR). pp 177-189 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In Russian)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

On the basis of the BOR-60 facility an analysis is made of characteristic requirements for safe operation of a fast-reactor power plant, from the standpoint of stability against perturbations in the operation of the main components and systems and against deviation from the design values of process parameters. The self-regulation properties of the facility are demonstrated, which ensure smooth transient regimes even when there are perturbations in reactivity and in the flow rate and temperature of the sodium at the reactor inlet. Regulation by neutron flux and by sodium outlet temperature are compared from the point of view of increasing the reliability of the control system and minimizing temperature strains in the components during transient processes. The calculated and experimental data on optimization of emergency shut-down and cooling regimes using different combinations of equipment are analyzed. The means and methods used to verify the state of the reactor vessel, fuel assemblies and in-vessel reactor systems during operation on power and during reloading are described; operating experience with sodium water steam generators is analyzed, as is the use of leak verification, control, and protection for steam generators under accident conditions.

- 4092 USE OF DEMONSTRATION FAST REACTOR

EXPERIENCE FOR ASSESSING THE RADIATION SAFETY OF FAST POWER REACTORS AT THE DESIGN STAGE. Bakumenko, O.D.; Kiselev, G.V.; Kulakovskii, M.Ya.; Pomerantsev, G.B.; Samarkin, A.A.; Sergeev, V.A.; Skorikov, N.V.; Tverdokskii, N.D.; Tsurukin, Yu.P. (Inst. of Physics and Power Engineering, Obninsk, USSR). pp 287-291 of Design, construction, and operating experience of demonstration LMFBRs. Vienna, Austria; International Atomic Energy Agency (1978). (In Russian)

From Symposium on design, construction and operating experience of demonstration liquid metal fast breeder reactors; Bologna, Italy (10 Apr 1978).

The results are presented of experimental studies on the behavior of fission products and radioactive corrosion products in the primary circuit of the BN-350 reactor during operation at power levels of up to 650 MW with defective fuel elements in the core. Data are also given on radiation conditions in the reactor building, dose burden to personnel and activity loss from the primary circuit. The experimental data thus obtained are compared with design data.

- 4093 CHEMICAL SIMULATION MODELING OF FUEL MOTION IN LMFBRs. Wheeler, P.A. Provo, UT; Brigham Young Univ. (1978). 146p. University Microfilms Order No. 79-11,877. Thesis (Ph. D.).

Chemical reactions were used to simulate fuel motion in LMFBR core disruptive accidents. The superficial vapor velocity necessary for transition between bubbly flow and dispersed flow was calculated for the chemical systems with the Kutateladze stability criterion. The actual vapor velocity at transition was measured experimentally in an open vessel. It was 50 to 100% higher than calculated. Closed vessel experiments showed a decrease in vapor velocity for a given concentration due to pressurization. A small heat sink added sufficient condensation to allow full dispersion of fluids even in a pressurized vessel. These results were compared to accident sequences in an LMFBR core.

- 4094 Opređenje temperatury v tvelakh yadernago reaktora (s zhidkometallicheskim teplositelem). (DETERMINATION OF THE TEMPERATURE IN FUEL ELEMENTS OF NUCLEAR REACTORS (WITH LIQUID METAL COOLANTS)). Sholokhov, A.A.; Zasorin, I.P.; Minashin, V.E.; Rumyantsev, V.N. Moscow, USSR; Atomizdat (1978). 232p.

An analysis is made of heat exchange in the active zone of a fast neutron nuclear reactor with a liquid metal coolant. Methods are given for computing temperature during changes in heat emission along the axis and cross-sectional direction of fuel elements. An examination is made of a method computing non-stationary temperature when there is a time change in the output of the reactor. A review is made of research results on heat exchange between the fuel and casing of the fuel element. A description is given of methods and results of computing temperature fields in fuel elements with local deterioration of heat exchange caused by fuel element defects. Considerable attention is given to a description of the physical aspect of the process. The subject material is illustrated by examples with numerical results. The book is intended for engineers engaged in the planning, construction, and operation of fast neutron nuclear reactors with liquid metal coolants, and for senior students in appropriate fields of specialization. 104 references, 60 figures, 23 tables.

- 4095 TEMPERATURE CONDITIONS IN AN LMFBR POWER PLANT FROM PRIMARY SODIUM TO STEAM CIRCUITS. Aubert, M.; Chaumont, M.; Moughnot, M.; Recolin, M. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France)); Acket, M. pp 305-310 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The subject is discussed under the headings: introduction; method and procedure (optimization program, review of process constraints - fuel limitations, turbine, steam generators, parameter selection - core parameters, temperatures, counterbalancing forces affecting temperatures); optimum values of parameters, with effect on costs; conclusion, (U.K.).

- 4096 OPTIMISATION OF A ONCE-THROUGH STEAM GENERATOR FOR LMFBR. Avanzini, P.G.; Barberis, M.; Monaci, R. (Nucleare Italiana Reattori Avanzati, Genoa). pp 349-353 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

Optimization studies are reported on the design of a steam generator consisting of a straight tube unit in a modular arrangement. The object was to determine the minimum total price of the unit assuming, as variables, materials tube size, matrix pitch fluids specific flow rates, losses of power, manufacturing and control costs, frequency of damage occurrence and repair times.

- 4097 OPTIMIZATION OF SODIUM PURIFICATION SYSTEMS FOR FAST REACTOR NUCLEAR POWER PLANTS. Baklushin, R.P.; Kozlov, F.A. (Institute of Physics and Power Engineering, Obninsk (USSR)). pp 383-389 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

A problem of rational construction of efficient primary and secondary sodium purification systems for fast reactor nuclear power plants is considered. On the basis of the BN-350 operating experience and published data some values are presented which allow evaluations to be made of the systems for the plants to come.

- 4098 OPTIMIZATION OF INTERMEDIATE HEAT EXCHANGERS FOR SODIUM COOLED FAST BREEDER REACTORS. Barratt, R.O.; Cox, J. (Foster Wheeler Energy Corp., Livingston, NJ (USA)); Beith, R.M.V. pp 355-367 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

Design evolution of the Intermediate Heat Exchanger for the U.S. Fast Breeder Programme is traced from the initial FFTF through to the Clinch River Designs and anticipated onward Commercial Plant Concepts. Supporting development work is outlined. Technical merits of certain features, such as sine wave tube

banks and replaceable tube banks (in-situ), are weighed up against quantified cost advantages with simpler arrangements. Future design trends are outlined which will lead eventually to fully optimised designs.

- 4099 U.S. SODIUM-COOLED BREEDER - SEEKING OPTIMIZATION. Beckjord, E.S.; Hardigg, G.W.; Iacobellis, S.F.; Laney, R.V.; Richards, R.B.; Squire, A. pp 23-29 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).
From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The subject is dealt with under the following headings: major objectives of the U.S. sodium-cooled breeder program; present status of program (evaluation of progress, FFTF status, CRBR status, PLBR status, study of proliferation resistant fuel cycles); recent studies of alternative design choices (loop-type systems, pool-type system, homogeneous vs. heterogeneous cores, steam cycle conditions, major components, refueling schemes, automated fuel fabrication process development); future plans.

- 4100 OPTIMISATION OF DECAY HEAT REMOVAL SYSTEMS FOR SNR 2 WITH RESPECT TO SAFETY AND RELIABILITY. Blombach, J.; Guenther, K.; Gyr, W. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp 37-42 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).
From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The general safety requirements for decay heat removal systems (DHR) are briefly described. The contradictory requirements of safety criteria, reliability of DHR and projected plant operation are shown. Two basic modes of DHR are identified, one using the main heat transfer loops, the other one being directly attached to the reactor vessel. Various system designs are possible for each mode, moreover combinations of the two basic models are feasible. Taking into account the safety requirements some systems prove to be inadequate. In addition it is shown that only combinations of different DHR systems can meet all requirements. For further optimisation the results of the reliability analyses of the SNR-300 DHR systems performed with respect to the licensing process are used. Taking into account the probability of common mode failures and the reliability target of DHR the necessary redundancy of each system is determined. A DHR concept is described covering both safety and reliability requirements.

- 4101 DEVELOPMENT OF FAST BREEDER REACTORS IN GERMANY, BELGIUM, AND THE NETHERLANDS. Brandsetter, A.; Eitz, A.W. pp 3-8 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The subject is treated in general and then under the following reactors: SNR-300, Superphenix, SNR-2. The SNR-2 is discussed in some detail: basic decisions, main boundary conditions, technical keypoints (primary system, reactor vessel, decay heat removal system, large components, containment, core design).

- 4102 COST/BENEFIT AND RISK/BENEFIT ANALYSES IN LMFBR PROGRAM PLANNING. Brewer, S.T.; Benson, R.A.; Palmer, R.S. pp 433-444 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The subject is discussed under the following headings: incentives analyses, uranium availability, electrical demand, the present value of future savings, alternatives to the breeder, environmental considerations, development program risks, results and conclusions.

- 4103 SIZING OF DECAY HEAT REJECTION LOOPS FOR LMFBRs. Broadley, D.; McSweeney, R.N. (Nuclear Power Co. (Risley) Ltd. (UK)). pp 117-124 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The paper briefly looks at the overall safety requirements for decay heat removal for LMFBRs. The number of DHR loops, and the way in which they are distributed, is not considered. The paper goes on to discuss the optimisation and sensitivity of the capacity of the DHR system, as a function of the peak shutdown reactor temperature. Calculations are performed for a pool-type CFR and for two designs of loop-type LMFBR. It is shown, as would be expected, that the relatively small thermal capacity of a loop-type reactor requires a relatively large DHR system to limit its temperature to an acceptable value.

- 4104 ANALYSIS OF SODIUM VOID COEFFICIENT IN HETEROGENEOUS CORE FAST POWER REACTORS. Bruna, G.B.; Cecchini, G.P.; Gastaldo, G. pp 91-100 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

In the framework of the Italian fast reactors program activities, neutronics studies were made on the heterogeneous core concept. The aim of the work was to single out some configurations and designs which permit a substantial improvement of the breeding and burn-up performance, in comparison to a homogeneous core, without affecting safety and, when possible, improving it. In the present paper, a summary of the main results obtained is presented, and some suggestions for the future development of optimisation activities are derived, in accordance with some assumptions determined by the Italian energy provisioning situation.

- 4105 ASSESSMENT OF THE EFFECT OF REACTOR SIZE ON HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS. Buttery, N.E.; Board, S.J. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). pp 57-65 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

There is a general tendency towards larger plant sizes, in the interests primarily of economies of scale. In this paper the effect

of core size on hypothetical core disruptive accidents (HCDA) is considered, and it is shown that the energy yield increases rapidly with size, primarily due to a tendency towards coherence of voiding in reactors with a large positive void coefficient. Small cores compare favourably in this respect with alternative large designs with low void coefficient cores, because the reduced mass more than compensates for the reduced doppler constant, and they also have a potential advantage in later stages of HCDA (transition phase and after). If energetic HCDA cannot be shown to be unrealistic and if containment of these events is provided as part of the general safety philosophy, then the costs (which may increase disproportionately with yield) of engineering an adequately reliable system needs to be accounted for. Containment costs are only one of many factors which need to be taken into account in optimising the design and so the energy release from a HCDA must take its proper place in the optimisation according to the safety principles and safety case agreed for LMFBRs.

- 4106 OPTIMISATION OF FUEL DESIGN IN RELATION TO REACTOR PERFORMANCE AND THE CONFLICTING DEMANDS OF OTHER PARTS OF THE FUEL CYCLE. Chamberlain, A.; Taylor, H.A.; Allardice, R.H.; Gatley, J.A. pp 175-179 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

An examination is made of the interactions between the designing of LMFBR fuel for reactor performance, encompassing generating costs and uranium conservation with the demands of the other parts of the fuel cycle notably fabrication and reprocessing. It is shown that the requirements are often conflicting and have to be weighed against each other to obtain an optimum balance whilst attaining an overall reconciliation with safety aspects.

- 4107 INTERMEDIATE HEAT EXCHANGER LEAK ON PHENIX PLANT AND THEIR REPAIR. Conte, F.; Sauvage, M.; Roumailhac, J.F. pp 291-296 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The incidents which have occurred concerning the PHENIX heat exchangers are discussed under the following headings; early years of PHENIX operation; action on the intermediate heat exchangers (description of heat exchanger, incident of 11 July 1976, incident of 3 October 1976, exchanger dismantling and repair, decontamination of exchangers, personal exposure); information gained from the incidents (safety of the plant, power station availability; information concerning the optimization of fast neutron power stations); conclusions.

- 4108 SUPER-PHENIX 1 FUEL ELEMENT OPTIMIZATION - FIRST CHECK OF OVERALL BEHAVIOUR. Delpeyroux, R.; Ollier, H.; Rousseau, J.; Peray, R.; Carteret, Y.; Dupouy, J.M.; Balloffet, Y.; Nervi, J.C. pp 125-131 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov

1977).

The subject is dealt with as follows: Super Phenix 1 fuel element characteristics and operating conditions; present methods of pin and clad behaviour evaluations; main options of the irradiation testing programme; tentative conclusions; Super Phenix 1 fuel pin definition.

- 4109 ENGINEERING ASPECTS OF HETEROGENEOUS AND HOMOGENEOUS REACTORS. Dickson, P.W. Jr. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). pp 269-278 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The core arrangement in an LMFBR can potentially affect the lower internals, upper internals, radial shielding, vessel, hot leg transients, head access area and control systems as well as breeding ratio, doubling time and core inventory. This paper describes the results of a study of the impact on these components and parameters that would result if the Clinch River Breeder Reactor were to incorporate a heterogeneous core.

- 4110 OPTIMISATION OF REACTOR DESIGN FOR NATURAL CIRCULATION DECAY HEAT REMOVAL IN A POOL-TYPE LMFBR. Durham, M.E. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). pp 67-76 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The paper presents the results of studies on the thermal-hydraulic behaviour of the primary circuit of a pool-type liquid metal cooled fast breeder reactor following the loss of A.C. supplies, in the event that natural circulation provides the only means of core heat removal. The effects of part load operation, and of providing secondary circuit cooling are considered, as well as some parameters associated with the design of the primary circuit. It is shown that optimisation of the primary pump rundown time alone is insufficient to ensure natural circulation. Substantial secondary circuit cooling is required during the primary pump run-down following operation at part-load to avoid the possibility of excessive temperatures in the core.

- 4111 EVOLUTION OF FRENCH FAST NEUTRON REACTOR CORE DESIGN AND PERFORMANCE. Estavoyer, M.; Ravier, J.; Chaumont, J.M. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France)). pp 189-196 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The subject is discussed with reference to the cores of Rapsodie, Phenix, Super Phenix 1 and Super Phenix II, under the headings: heterogeneous vs. homogeneous core; mechanical aspects of core design (reactivity, handling); subassembly design; core protection and monitoring.

- 4112 RELIABILITY TECHNOLOGY IN LMFBR OPTIMISATION. Fawcett, S. (UKAEA Reactor

Group, Culcheth). pp 85-90 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

Quantification and in-depth analysis, as used in safety and reliability assessments, are examined in four widely dissimilar applications. At the most general level, cost and effort for more intensive design, development and reliability programmes to achieve higher availability are optimised against extra revenue, but the scope for quantification is, as yet, limited. Both safety and availability criteria apply in optimising structural design, development and manufacture: progress in quantification and analysis is observed on responsive aspects. Introducing availability into optimising the number and size of cooling circuits (and also turbine generating plant capacity) might call for more spare capacity in LMFBRs than has previously been needed in power stations.

- 4113 CDFR FUEL DESIGN OPTIMISATION. Gatley, J.A.; Brindley, K.W. (Nuclear Power Co. (Risley) Ltd. (UK)). pp 167-173 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The main considerations that have been involved in progressing from generalised parametric surveys to the determination of the leading features of a preferred fuel design for CDFR are described. The criteria of merit and practical engineering limitations are discussed, and detailed results of studies of fuel pin diameter, and wrapper and fuel can thickness are presented.

- 4114 STEAM CYCLE FOR HIGH POWER LIQUID METAL FAST BREEDER REACTOR PLANTS. Gaudy, B. (Electricite de France, 92 - Courbevoie. Service Etudes et Projets Thermiques et Nucleaires). pp 297-303 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

Out of the various types of possible steam cycles, it is proposed to select three of them: the reheating cycle by sodium, the reheating cycle by extracted steam with moisture separation, the reheating cycle by extracted steam without moisture separation.

- 4115 THERMODYNAMIC AND FLUIDDYNAMIC ASPECTS IN OPTIMIZING THE DESIGN OF FAST REACTOR SUBASSEMBLIES. Hoffmann, H.; Weinberg, D. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). pp 133-139 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

Temperature distributions and pressure drops were calculated for sub-assemblies with grid-type and helical spacers by means of experimentally supported codes. The influences of rod arrangements and power form factors were investigated. Optimum bundles with grid-type spacers were selected for which the temperature

form factor reached a minimum. All considered bundle configurations are optimized for identical maximum coolant temperatures and identical pressure drops. An optimum subassembly can then be found with respect to the overall thermal behavior of the plant.

- 4116 GENERAL ELECTRIC/BECHTEL PROTOTYPE LARGE BREEDER REACTOR (PLBR). Horst, K.M.; Gibson, A.S.; Davies, S.M. (General Electric Co., Sunnyvale, CA (USA)); Golan, S. pp 223-237 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The General Electric/Bechtel PLBR is a conservatively designed 1000 MWe gross (914 MWe net) plant which gives confidence that a practical LMFBR can be built for U.S. application. A significant finding is that the LMFBR achieves lowest costs with a low reactor outlet temperature (468°C) and a low pressure (6.89 x 10⁶ Pa) saturated steam cycle. Moreover, the risks of failing to meet performance or other goals are lower for these cycle conditions than for higher temperature superheat cycle conditions.

- 4117 SODIUM DEPOSITION FROM FAST REACTOR GAS BLANKETS - FACTORS INFLUENCING THE CHOICE OF ARGON OR HELIUM COVER GAS. Hotchkiss, R.C.; Roberts, R.A. (Central Electricity Generating Board, Southampton (UK). Marchwood Engineering Labs.). pp 147-157 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

Sodium evaporates from the sodium/gas interface in fast reactors. Cooling of sodium laden gas due to thermal gradients in the gas space above the sodium results in the formation of an aerosol. Sodium condenses directly from the vapour phase and the aerosol deposits of the cooler surfaces exposed to the cover gas. The major difference between argon and helium cover gas is the dependence of density of the sodium saturated gas upon temperature. The density of argon displays the usual inverse dependence upon temperature but that of sodium saturated helium at reactor hot pool temperatures is almost independent of temperature because of the low molecular weight of the gas compared with the evaporating species. Convection is therefore the main sodium transport mechanism in argon but diffusion is more important in some situations with helium. Simple comparative experiments on evaporation and deposition of easily handled liquids in argon and helium are described. Apparatus has been constructed to measure sodium deposition rates from argon and helium cover gases under conditions relevant to a fast reactor. The results are compared with a natural convection mass transfer theory which describes the evaporation of sodium from the pool surface under conditions of aerosol formation. Sodium loadings have been measured in helium (and should soon be obtained in argon). Attempts have been made to estimate the aerosol size distribution by catching the particles on fine fibres and examining them in a scanning electron microscope. Area integrated sodium deposition rates, which are approximately equal to the evaporation rates, in argon are close to calculated open pool evaporation rates. Integrated rates in helium are considerably lower than in argon but local deposition rates at the top of the cover gas

space, which is the region of most concern, may not be significantly lower. Local deposition rates from He indicate that deposition is much less uniform than from Ar.

- 4118 MECHANICAL DESIGN TRADE-OFF STUDIES OF REACTOR CORE FOR LMFB. Iwayoshi, S.; Hasegawa, Y.; Ozawa, M.; Iwai, S. pp 141-146 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

This paper discusses the relationships between concepts of the core support system, fuel handling system and reactor shutdown system, which poses major problems in the design of the reactor internals for future LMFBs. Designers of the LMFB reactor internals will be forced to solve the structural materials problems which are caused by the fast neutron irradiation. The enhanced bowing of the core component ducts due to radiation-induced swelling has posed difficult problems in the core restraint design. Three types of core restraint systems - the active, passive and the static systems - were compared with each other along with such additional considerations as the acceptable load capacity of IVHM, the alignment of the absorber assembly channels and the controlling reactivity and the maintainability of the structure elements of the active system. The fluence limit requirements on core support structures to minimize the irradiation-induced degradation of the mechanical properties of the structural materials demanded the choice of the design concept which incorporates replaceability of any structure elements essential to safety in order to avoid longer fuel assembly features caused by the provision of considerably thick shielding region. But this option brought the restriction in available in-vessel fuel handling systems. The conceptual studies of absorber assemblies and drive mechanisms, especially scram assisting methods, for the primary and the secondary shutdown systems have also been conducted by taking account of such effects on scram times as bowing of the individual control duct assemblies, displacements caused by core deflection and earthquakes.

- 4119 DESIGN CONSIDERATIONS FOR THE PRIMARY SYSTEM AND THE PRIMARY COMPONENTS OF SNR-2. Kohler, M.; Hillekum, H.; Ratzel, W.; Wagner, J.; de Boer, G.; Heslenfeld, M.W. pp 317-323 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The conceptual design of the primary system, including its large components, of the 1300 MWe SNR-2 demonstration plant is described. As the SNR-300 Kalkar plant, the SNR-2 is based on the loop concept. Four heat transfer loops, each arranged in a cylindrical primary cell (pot) close to the reactor vessel, are foreseen. The atmosphere inside the primary cells is heated to the temperature of the primary sodium. Inside each cell, 2 IHX's, 1 hot leg pump, 2 valves and the piping, support and instrumentation structures are arranged. The pot is the common cavity in case of leaks.

- 4120 OPTIMIZATION STUDY OF A DEMONSTRATION FAST BREEDER REACTOR PLANT. Kurosawa, K.; Sugizaki, T.; Kataoka, I.; Kishida, K.

(Mitsubishi Atomic Power Industries, Inc., Omiya, Saitama (Japan). Engineering and Development Div.); Tomabechi, K.; Nakai, Y.; Miki, R. pp 15-22 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

An optimization study on the reactor heat transport system of a 1,000 MW e loop type fast breeder reactor plant has been conducted. This study includes survey of the main coolant pipe size, the number of loops and sizing and selection of main components. Steam cycles, steam conditions and the system operating temperatures are also investigated. Normalized electrical generation cost comparison is made for various combinations of these plant parameters.

- 4121 DESIGN CONSIDERATIONS AND COST ANALYSES OF THE SECONDARY SYSTEM OF SNR 2. Lievense, K.; Jansing, W.; Vinzens, K.; de Hes, M.; Raas, J.L.; Renders, A.H.I.; van Westenbrugge, J.K. pp 311-316 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

A review is given of the considerations influencing the concept for the SNR 2 secondary sodium loops, including the steam generator buildings and the steam generators.

- 4122 FAST REACTOR CORE RESTRAINT SYSTEMS. Loveday, S.F.; Beckett, V.S. (Nuclear Power Co. (Whetstone) Ltd. (UK)). pp 181-188 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The normal operational and transient behaviour of the preferred restrained core design for a commercial size fast breeder reactor is described and its performance illustrated by typical analytical results from purpose written computer codes. Two alternative design concepts are considered and their relative advantages and disadvantages highlighted. The preferred design of restrained core employs a barrel type structure around the outside of the core shield elements. This structure restrains the core via contact pads on each sub-assembly at two levels; the lower just above the active core region, and the upper just below the tops of the sub-assemblies.

- 4123 STUDY OF ALTERNATIVE BOILER FEED PUMP ARRANGEMENTS FOR A CIVIL FAST REACTOR. Marshall, F. (UKAEA Reactor Group, Culcheth). pp 325-331 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The paper assumes a design utilizing two standard 660 MW Turbo Generators and examines various feed pump arrangements to suit such a scheme. The paper predicts the performance of a combination of steam and electric driven pumps compared with all electric driven pumps and suggests that there will be a distinct advantage in adopting an all electric system.

4124 ATOMICS INTERNATIONAL (AI) PROTOTYPE LARGE BREEDER REACTOR (PLBR). McDonald, J.S. (Atoms International Div., Canoga Park, CA (USA)); Campise, A.V.; Brunings, J. pp 239-246 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The AI-PLBR breeder plant design prepared for ERDA and EPRI is of 1000 MWe size, utilizing a loop-type sodium system configuration and producing 2200 psig/850°F steam. A 'bullseye' core geometry type sodium system configuration and is employed with PuO₂-UO₂ fuel and UO₂ fertile material. The reactor outlet coolant temperature is 930°F. A modified 'A'-frame refueling system is employed, which is capable of handling 1/3 of a core loading in 12 days. An inducer-type mechanical pump is used in the primary circuit because of its excellent NPSH characteristics. Hockey sticks steam generators are used to produce near-fossil steam conditions at the turbine throttle. The balance of plant (BOP) design was developed by the architectural-engineering firm of Burns and Roe. It includes Allis-Chalmers - KWU 1800-rpm tandem-compound turbine, selected with high-, intermediate-, and two double-flow low-pressure cylinders equipped with two moisture separator cyclones. A design feature to enhance the licensability of the reference design is the three-level Decay Heat Removal System (DHRS), which consists of normal, backup, and diverse decay heat removal paths. The Atomics International PLBR design illustrated the technical soundness of the LMFBR system for meeting the world's long-term electrical energy supply needs. The design includes design features to assure licensability and innovative engineering features to enhance reliability, constructability, economics, and U.S. utility grid compatibility. The design concept provides a sound basis for the future detailed design of a prototype plant and subsequent development of larger LMFBR plants.

4125 INCORPORATING THE UTILITY VIEWPOINT IN THE DESIGN OF THE PROTOTYPE LARGE BREEDER REACTOR (PLBR). Minnick, L.E. (Electric Power Research Inst., Palo Alto, CA (USA)). pp 207-210 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The PLBR is intended to become the first full-scale demonstration breeder plant in the U.S.A. Three reactor designer/architect engineer teams have been developing plant designs under the joint sponsorship and management of ERDA and the Electric Power Research Institute. The evaluation of many difficult trade-offs is implicit in the design of such an initial demonstration plant. In general, utility requirements emphasize conservative design, use of demonstrated technology to the extent possible, minimum new development, maximum anticipated reliability, minimum complexity, and the lowest predicted operating costs possible in view of the stated considerations.

4126 LMFBR STEAM CYCLE DESIGN TRADE-OFFS. Morabito, J.J.; Ross, F.A. pp 369-382 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of

sodium cooled fast reactors; London, UK (28 Nov 1977).

Large breeder reactor plant conceptual design studies performed by three reactor manufacturer-architect engineer teams resulted in the recommendation of two alternative steam cycle conditions and associated reactor coolant outlet temperatures: a superheated cycle, supplying steam at 2200 psig (15,168 k Pa), 850°F (454°C) to the turbine; and a dry and saturated cycle, supplying steam at 985 psig (6,791 k Pa), 545°F (285°C) to the turbine. Extensive evaluations of the comparative attributes of these alternative steam cycle conditions were performed to facilitate a choice. The key issues and comparative cycle attributes identified, and the conclusions of these evaluations are outlined in this paper.

4127 OPTIMIZATION STUDY OF LMFBR HAVING SMALL SODIUM VOID COEFFICIENT FROM FUEL CYCLE PERFORMANCE VIEWPOINT. Nishimura, H.; Yoshida, H.; Hirata, M. pp 43-48 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

Under the constraint of an allowable level of sodium void coefficient in a mixed oxide fuel 1000MWe LMFBR, three types of core concepts - a pancake core, an annular core and an internal blanket concept - were investigated on the basis of same analysis methods. The twenty-five group JAERI-Fast set was used as the basic cross section data. Neutronics calculations containing sodium void coefficient predictions and burnup characteristic analyses were performed based on the two-dimensional diffusion method and six-energy group cross sections collapsed. The fuel cycle performance was analyzed on such items as burnup characteristics, breeding performance and fuel inventory. It was found that the annular core concept had a flexibility to a more strict constraint upon the sodium void coefficient.

4128 EFFECT OF CORE DESIGN CHANGES ON THE DOUBLING TIME AND THE FUEL CYCLE COST OF A 1,000 MWE LMFBR. Otake, I.; Inoue, T.; Tomabechi, K. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)); Osada, H.; Aoki, K. pp 409-418 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

Core design studies were performed to improve the doubling time and to minimize the fuel cycle cost of a 1,000 MWe Fast Demonstration Reactor. A core was designed mainly based on the technology being used for the design of a prototype fast reactor MONJU, because much valuable experience will be forthcoming from this reactor. Design parameters with a wide variable range were used to clarify the relations between breeding characteristics, fuel economics and various designs.

4129 EFFECT OF STEAM CYCLE CONDITIONS UPON THE ECONOMICS AND DESIGN OF A SODIUM-COOLED FAST REACTOR. Philpott, E.F.; Pounder, F.; Wilby, C.R. (Nuclear Power Co. (Risley) Ltd. (UK)). pp 333-348 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov

1977).

The paper studies the effect of variation of steam and feedwater conditions upon the economics, design and layout of a sodium-cooled fast reactor. The parameters investigated are steam temperature and pressure, feedwater temperature, and boiler recirculation ratio. The paper also includes an assessment of the effects of associating the fast reactor with saturated steam cycle conditions.

- 4130 HOW THE ARCHITECT-ENGINEER MANAGES DESIGN OBJECTIVES AND RESTRAINTS FOR OPTIMIZING SODIUM COOLED REACTORS. Roe, K.A.; Roe, K.K. (Burns and Roe, Inc., Oradell, NJ (USA)). pp 197-205 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The design objectives of low capital and operating costs and high reliability are best attained by carefully defining criteria early in the development stage. Throughout the design development, unusual attention to constructibility, reliability and availability requirements, and the early resolution of licensing issues by designated engineering specialists are some of the approaches used to minimize design restraints. Effective management of these design objectives and restraints can assure that, on balance, LMFBR costs can be improved, reliability increased, and maintenance can be effective.

- 4131 DESIGN TRADE-OFFS IN VIEW OF SAFETY CONSIDERATIONS. Saji, G.; Kishida, K. (Mitsubishi Atomic Power Industries, Inc., Tokyo (Japan)); Inoue, T. pp 49-56 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

In view of resolving conflicting demands of cost, safety, flexibility of operation and design margins, safety design of various plant systems is discussed referring to their weight on construction costs. An influence of hypothetical core disruptive accident (HCDA) and loss of piping integrity (LOPI) on plant design and thus on construction materials is discussed, in optimising future commercial FBR plants.

- 4132 LOOP AND POOL FAST REACTORS. Taylor, D. (Nuclear Power Co. (Risley) Ltd. (UK)); Cage, F. pp 211-222 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

There are basically two different ways of arranging the primary circuits of L.M.F.B.R.'s. One in which all the components are housed in separate vessels connected together by pipework, normally referred to as the loop and the other whereby all the components are housed in a single large vessel, normally referred to as the pool. The paper discusses the two arrangements and identifies some of the major engineering differences and the effect these have upon design and operation.

- 4133 RECENT COMMERCIAL FAST REACTOR DESIGNS WITH IMPROVED SAFETY FEATURES. Thornton, D.E.J. (UKAEA, Risley. Nuclear Power

Development Establishment); Brindley, K.W. pp 77-84 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

Most CDFR design work in the UK in the past has related to simple cylindrical cores of fuel surrounded by breeder material. It became apparent, however, that with this design there was room for improvement in the response to certain hypothetical accidents in which sodium coolant was expelled from the core, and a number of alternative designs have been examined in recent years. Most of these variants involved simple alterations in core layout using existing subassembly designs, principally by introducing local breeder regions into the fuelled core, in the form of radial spokes, annuli or clusters. A few others modified the subassembly design, using axial shortening, introduction of breeder layers, and reducing the sodium fraction; the effect of reducing the core size was also examined.

- 4134 OPTIMIZATION STUDY OF LMFBR WITH THORIUM-BLANKET FOR THE SUPPLY OF U-233 TO MULTI-PURPOSE HTGR. Yoshida, H.; Nishimura, H.; Hirata, M. (Japan Atomic Energy Research Inst., Tokai, Ibaraki. Tokai Research Establishment). pp 401-407 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The conventional, parfait blanket and heterogeneous core LMFBR concepts with thorium-blanket have been investigated to identify potential configurations of fast breeder reactor for the supply of U 233 to the associated multi-purpose HTGR in the symbiotic system, from the view points of fuel sufficiency, inherent safety and fuel cycle cost. It was found that the parfait concept could be preferable as the LMFBR in the symbiotic system from its best performances in the condition imposed.

- 4135 INHERENTLY SAFE REACTOR DESIGN. Lancet, R.T.; Specht, E.R.; Cooper, M.; Adkins, R.; Bailey, H.S.; Grantz, A.L.; Millunzi, A. pp 101-110 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The U.S. LMFBR Inherently Safe Reactor Design Program is discussed with the emphasis on the results obtained to date and future plans. Specific design concepts for inherent core shutdown systems are presented as well as the corresponding experimental verification program.

- 4136 CONSEQUENCES OF SAFETY PRESCRIPTIONS FOR FAST BREEDER REACTOR DESIGN IN FRANCE. Megy, J.M. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)); Cravero, M.; Leduc, J.; Noel, H. pp 9-13 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

An approach to optimization of a power plant and its boiler is described, with particular

reference to Rapsodie, Phenix, Super Phenix and Super Phenix 2, under the following headings: accident prevention (at the option stage: through component reliability; core cooling; reactor shut-down; cooling system failure detection); accident containment (earthquakes; missiles; sodium fires; core fusion accidents).

- 4137 PHYSICS AND SAFETY ASPECTS OF PRELIMINARY SNR 2 CORE DESIGNS. Morelle, J.M. (Internationale Natrium-Brutreaktor-Bau G.m.b.H., Bergisch Gladbach (Germany, F.R.)); Evrard, G.; Renard, A.; De Wouters, R.; Pilate, S.; Wehmann, U.; Spenke, H.; Loehr, H.; Kieflhaber, E. pp 31-36 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

Recent design work on SNR 2 has involved inserting fertile regions into inner parts of the core, to improve inherent safety characteristics, to improve the breeding capabilities and to improve the core performance. These aspects are discussed.

- 4138 SAFETY OF NUCLEAR POWER PLANTS WITH FAST BREEDER REACTORS - ENGINEERING ASPECTS. Adam, E.; Schwarz, M. (Technische Univ., Dresden (German Democratic Republic). Sektion Energieumwandlung). Wiss. Z. Tech. Univ. Dresden; 27: No. 6, 1305-1309(1978). (In German).

For the safe operation of nuclear power plants with fast breeder reactors the removal of energy released under operational or accidental conditions is important. Proceeding from considerations on the basis accident 'total voltage breakdown', two cooling systems (loop construction and pool construction) are studied for a sodium-cooled reactor. Criteria for the efficiency of cooling systems are the temperatures within the primary coolant circuit after an accident. The temperature curves have been calculated, using a computer code. The results obtained are discussed and conclusions are drawn for designing the whole nuclear power plant.

- 4139 POSSIBLE LONG TERM OPTIONS FOR THE FAST REACTOR PLUTONIUM FUEL CYCLE. Flowers, R.H.; Johnson, K.D.B.; Miles, J.H.; Webster, R.K. (United Kingdom Atomic Energy Authority, Harwell, England). pp 256-263 of Energy technology V: challenges to technology. Hill, R.F. (ed.). Washington, DC; Government Institutes, Inc. (1978).

From 5. energy technology conference; Washington, DC, USA (27 Feb 1978).

The role of fuel reprocessing in a fast reactor fuel cycle is discussed with reference to an advanced point in time at which fast reactors dominate the nuclear power base. Emphasis upon minimisation of the out-of-reactor fuel inventory, and the consequent higher activity levels present in the reprocessing plant, may lead to a progressive introduction of co-processing, gel fuel production, and integrated fuel reprocessing-fabrication plants. The further possibility of a relaxation in the degree of fission product removal in a remotely operated integrated plant is introduced, and conceptual flowsheets to achieve particular levels of fission product recycle are described.

- 4140 OPTIMIZATION OF THE WESTINGHOUSE/STONE AND WEBSTER PROTOTYPE LARGE BREEDER REACTOR.

Anderson, C.A. Jr. (Westinghouse Electric Corp., Pittsburgh, PA (USA). Advanced Reactors Div.). pp 247-259 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

The optimization of the Westinghouse/Stone and Webster Prototype Large Breeder Reactor (PLBR) is described. This reactor plant, designed for ERDA and EPRI, resulted from optimization and tradeoffs on plant size, number of loops, steam cycles, system temperature, pump location, refueling concept, reactor shut-down system logic, control system logic, steam generating system, residual heat removal system, core arrangement and reactor vessel. The result is a three loop LMFBR rated at 1000 MWe gross with sodium entering the reactor vessel at 650°F (345°C) and leaving at 950°F (510°C). The reactor vessel has a flat closure head on top, containing three rotating plugs on which are mounted the upper internals structure and the in-vessel transfer machine. The core has three radial layers of core material separating four radial blanket regions. An inclined refueling chute penetrates the reactor vessel. Plant efficiency of 37% is achieved with the use of once-through steam generators operating in the modified Sulzer mode, producing 2200 psi, 850°F (15.2 x 10⁶ pa/455°C) steam. The residual heat removal system (RHRS) consists of three independent heat removal paths in which the intermediate sodium is cooled in air blast heat exchangers (author).

- 4141 POTENTIAL OF LARGE HETEROGENEOUS REACTORS. Barthold, W.P.; Tzanos, C.P.; Beitel, J.C. (Argonne National Lab., IL (USA)). pp 279-290 of Optimization of sodium-cooled fast reactors. London; Thomas Telford for the British Nuclear Energy Society (1978).

From BNES conference on optimization of sodium cooled fast reactors; London, UK (28 Nov 1977).

Different 1200 MWe heterogeneous core configurations have been evaluated and compared with a homogeneous 1200 MWe LMFBR design. This study discusses the neutronic coupling concept which was employed to discriminate between different heterogeneous core designs. Reactivity control, breeding performance, safety parameters, fuel pin optimization, transient behaviour and thermal performance were investigated in detail and results of these analyses are presented.

- 4142 STUDIES ON MOLTEN FUEL. Goetzmann, O.; Schulz, B. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung); Dorner, S.; Fieg, G.; Werle, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik); Alsmeyer, H.; Barleon, L.; Guenther, C. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorbauelemente); Perinic, D. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Bereich Ingenieurtechnik). KFK Nachr.; 10: No. 3; 4, 87-91(1978). (In German).

In building the SNR 300 it was assumed that melting of the whole core had to be dealt with and that the proof of retention within the tank could not be furnished in time for the SNR 300. In order to cover the risk associated with this there was provided an external core-catcher which can be cooled and which is coated by UO₂ resp. ThO₂. The work carried out with respect to core meltdown is oriented on this problem

and comprises investigations in the following areas: - Model experiments and theoretical studies on thermodynamics. - Explanation of important phenomena in the interaction of reactor materials (UO₂, steel, Na) at representative temperatures. - Determination resp. confirmation at representative temperatures of materials data required for transferring the thermodynamics results to core meltdown. - With respect to the work concerned with gas-cooled breeders, a core-catcher concept for this reactor type.

- 4143 EXPERIMENTAL STUDIES ON LOCAL COOLING PERTURBATIONS. Huber, F.; Peppier, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung); Casal, V. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorbaulemente). KFK Nachr.; 10: No. 3; 4, 101-104(1978). (In German).

The following statements can be made and applied to a SNR Mk Ia cluster: - Under SNR 300 nominal conditions the boiling temperature will only be reached behind wall or corner obstructions of greater than 15% of the blocking cross-section. - Without the influence of gas the region of the obstruction itself can be cooled relatively well even under boiling conditions. The interval between the conditions at the beginning of boiling and those leading to continuing interruption of cooling depends on the form of boiling that will occur. - If in combination with an obstruction there will be released fission gas direct transition from the single phase stage to local interruptions of cooling will be possible. This statement is true, however, only for the region immediately behind the obstruction. - Direct transition from local boiling to boiling in the whole cluster, caused by thermodynamical instability, can be excluded.

- 4144 IN-PILE INVESTIGATION OF LOCAL COOLING PERTURBATIONS (MOL 7C). Kramer, W.; Schleisiek, K. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung); Gelthoff, D.; Schmidt, L. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung); Vanmassenhove, G.; Vervimp, A. (Centre d'Etude de l'Energie Nucleaire (Belgium)); Rahn, A. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)). KFK Nachr.; 10: No. 3; 4, 135-108(1978). (In German).

In the BR2 in Mol there was to be studied in three experiments the behavior of a sodium-cooled fuel subassembly under locally disturbed cooling conditions. The present paper gives a report on the kind of performance and the results of the first two experiments. The third one is planned for the end of 1979.

- 4145 VALIDATION OF THE COMPUTER CODE ARES FOR CALCULATING THE TANK LOAD UNDER OVERPOWER CONDITIONS. Malmberg, T.; Hoang, Y.S. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). KFK Nachr.; 10: No. 3; 4, 83-86(1978). (In German).

The present paper deals with the fundamental aspects of the computer code ARES as well as the experimental set-up and its simulation by the code. In addition the occurring hydromechanical phenomena are described qualitatively. Finally experiment and calculation are compared with one another.

THORIUM FUELS IN POWER REACTOR SYSTEMS. Nair, S. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Ann. Nucl. Energy; 5: No. 6, 259-266(1978).

A study has been made of the relative hazards of waste arising from the use of thorium fuels in certain power reactor systems. The hazard estimates were made using the ICRP maximum permissible concentrations for the ingestion pathway for both water-soluble and insoluble wastes. Two separate conceptual fuel cycles were considered, the one (referred to as the advanced fuel cycle) involving the use, in the THTR, of thorium and uranium bred in an LMFBTR thorium blanket, the other involving the use of thorium and ²³⁵U-enriched uranium in a THTR. The waste hazards were compared with those resulting from the use of ²³⁵U-enriched fuel in HTR and PWR and the effect of recycling uranium and ²³⁹Pu-rich plutonium was also investigated. Thorium, uranium and plutonium extraction efficiencies of 99.9% were assumed wherever relevant before waste hazard estimates were computed.

- 4147 PHOTOELASTIC INVESTIGATION OF A REACTOR BUILDING. Jullien, F.J.; Morel, A. pp 855-859 of 6. Internationale Konferenz 'Experimentelle Spannungsanalyse'. Duesseldorf, Germany, F.R.; VDI-Verl. (1978). (In French)

From 6. conference on experimental stress analysis; Munich, F.R. Germany (18 Sep 1978).

This paper is devoted to the study of the nuclear reactor containment sheltering the Super Phenix Surgenerator. This structure of civil engineering is very important because of its sizes and of its complexity. A model of this building and of the next soil has been fabricated with a reduced Scale of 1/200; the soil has been there considered as a boundary condition. The fabrication of a closely fitting model has required many studies on the materials, and on the technology adapted to the mouldering and the loading of this reduced structure. The results confirm the whole behavior as obtained by the calculation. In addition, they complete the study by giving the local behavior in singular areas; in this way, they allow the optimisation of the design.

- 4148 OPERATING EXPERIENCE WITH THE PFR EVAPORATORS. Evans, A.D.; Broomfield, A.M.; Smedley, J.A.; Bray, J.A. (UKAEA, Dounreay. Nuclear Power Development Establishment). pp 3-15 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

This paper describes operating experience with the PFR evaporator units with particular emphasis on those features associated with the use of 2 1/4 Cr 1Mo steels. The problems of small leaks in the tube to tube sheet welds, where twelve out of the three thousand used have been affected, are discussed. Normal operation of the units has demonstrated that the use of 2 1/4 Cr 1Mo steel is entirely satisfactory for both the sodium and water circuits involved.

- 4149 DESIGN PHILOSOPHY AND FUNCTIONAL REQUIREMENTS OF A SODIUM HEATED STEAM GENERATOR MADE OF FERRITIC STEEL. Robin, M.G.; Birault, J. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Div. d'Etude et de Developpement des Reacteurs). pp 50-54 of

Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

Since the first use of 2.25 Cr-1Mo Croloy for the first single wall sodium heated steam generator installed, in the early sixties, on the Enrico Fermi Atomic Power Plant, it has been recognised that ferritic steels would be suitable for such applications. Unfortunately, due to decarburization in liquid sodium, and low creep strength, 2.25 Cr-1Mo steel is somewhat limited in its operational temperature range. Starting from the design philosophy of a once-through unit and considering the functional requirements of a steam generator operated under normal and abnormal conditions, it is shown that, after suitable development, a high strength 9% Cr ferritic steel could be used for the tube bundle of a modern high temperature and pressure steam generator for the LMFBR.

- 4150 SODIUM HEATED STEAM GENERATORS. NEAR TERM AND PROJECTED INFORMATION NEEDS-FERRITIC STEELS. Spalaris, C.N.; Challenger, K.D.; Dutina, D.; Ring, P. (General Electric Co., San Jose, CA (USA)). pp 55-63 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

The use of 2-1/4Cr-1Mo as steam generator material in the U.S. was predicated upon the assumption that consumable remelting be used for processing tubing and tubesheets. This was necessary to assure high quality sodium/water pressure boundary members and tube-to-tubesheet welds, envisioned as decisive factors for achieving high steam generator performance potential. Recognizing that the remelting processes may alter the behaviour of the alloy, development programs are being carried out to confirm that ASME Code design requirements for 2-1/4Cr-1Mo can be met. Additionally, environmental effects such as corrosion and decarburization are being studied to assure that data base from air melted steel used in the development of our present understanding of the alloy can be applied in the design of components. Cost differentials for consumable remelting have been modest in comparison to the cost of finished steam generator modules. On-going development programs are addressing most identified areas of concern, although specific requirements may arise as more experience is gained through operation of ferritic sodium heated steam generators.

- 4151 POST-PERFORMANCE EXAMINATION OF THE 50 MW STEAM GENERATOR FOR THE PROTOTYPE FAST BREEDER REACTOR 'MONJU'. Tsuda, H.; Kawara, M.; Shibato, E. (Hitachi Ltd., Ibaraki (Japan). Hitachi Works); Kanamori, A. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). pp 22-26 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

This paper describes the disassembly and examination of the 50 MW Steam Generator

following performance tests at the 50 MW Steam Generator Test Facility. This disassembly and examination consists of a visual examination, dimensional check and nondestructive examination of parts of interest, metallurgical examination of tubes and so forth. On the basis of the results of this examination, it was concluded that the integrity of the 50 MW Steam Generator was confirmed and that there was negligible degradation of materials or components during the performance test.

- 4152 RESULTS OF THE TEST AND INSPECTION PROGRAMME OF THE PROTOTYPE SNR-300 STEAM GENERATORS. Van Westenbrugge, J.K.; Tromp, Th.J.; De Maeijer, J.R. (Neratoom N.V., The Hague (Netherlands)). pp 16-21 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

Results are presented of the dismantling and inspection of two of the Neratoom prototype steam generators. Neither the straight tube nor the helical coil tube steam generator showed major problems.

- 4153 MATERIAL CHOICES FOR THE COMMERCIAL FAST REACTOR STEAM GENERATORS. Willby, C.; Walters, J. (Nuclear Power Co. (Risley) Ltd. (UK)). pp 40-49 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

Experience with fast reactor steam generators has shown them to be critical components in achieving a high availability. This paper presents the designers views on the use of ferritic materials for steam generators and describes the proposed design of the steam generators for the Commercial Fast Reactor (CFR), prototype of which are to be inserted in the Prototype Fast Reactor at Dounreay.

- 4154 BREEDER: NOW OR NEVER. Murphy, P.M. (Electric Power Research Inst., Palo Alto, CA). Proc. Am. Power Conf.; 40: 86-91(1978). (CONF-780440--).

From American power conference; Chicago, IL, USA (Apr 1978).

The timing of the commercial introduction of the liquid metal fast breeder reactor (LMFBR) will be an important factor in its ability to supply a significant fraction of the nation's future electrical needs. The number of breeders we can build initially will be limited by the size of our low-cost uranium resources and by the rate at which light water reactors (LWRs) are placed in service. Since this uranium resource is fixed in size while electrical demand will grow geometrically, it is clear that the sooner the breeder is introduced commercially the larger will be the fraction of electrical demand it can supply. An early commercial introduction on an adequate scale requires full-scale resumption of LWR construction and redirection of LMFBR development programs toward a near-term commercial prototype.

- 4155 HISTORIC DEVELOPMENT OF STEELS CONTAINING 8-10% MO AND FUTURE DIRECTIONS FOR

LMFBR APPLICATIONS. King, R.T. (Oak Ridge National Lab., TN (USA)); Egnell, L. (Sandvik AB (Sweden)). pp 74-81 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

Recent developments of steels containing about 8-10% Cr and 0-2% Mo are reviewed for LMFBR steam generators and structural applications. Several promising alloy compositions are identified. Future research to qualify the better materials for service must further define effects of processing history and chemistry including heat treatment, and move in an orderly manner to acquisition of long-term data and demonstration of test components in prototypic coolant loops.

- 4156 DEVELOPMENT OF 9CR-MO STEEL FOR FAST BREEDER REACTOR STEAM GENERATORS. Yukitoshi, T.; Yoshikawa, K.; Tokimasa, K.; Kudo, T.; Shida, Y.; Inaba, Y. (Sumitomo Metal Mining Co. Ltd., Ichikawa, Chiba (Japan). Central Research Lab.). pp 87-90 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

Newly developed 9Cr-2Mo steel has not only good formability and weldability, but also higher allowable stresses than 2-1/4Cr-1Mo or 9Cr-1Mo steels. This new steel has already been used without trouble for over 2 years as reheater and superheater-tubes in boilers. This paper gives a summary review of the basic characteristics such as mechanical properties, creep properties, fatigue properties, weldability and corrosion resistance which will be required when 9Cr-2Mo steel is applied to LMFBR steam generators.

- 4157 MATERIALS DATA REQUIREMENTS FOR THE HIGH-TEMPERATURE STRUCTURAL DESIGN OF 2 1/4CR-1 MO STEEL COMPONENTS. Corum, J.M. (Oak Ridge National Lab., TN (USA)); Dalcher, A.W. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). pp 69-73 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

The materials data needs that can arise when a structural alloy is used for the first time in liquid-metal fast-breeder reactor system components are discussed from the standpoint of the high-temperature structural designer. Recent experience with 2 1/4 Cr-1 Mo steel is used as an example. A common structural problem, involving repeated thermal transient loadings, is discussed, and the current high-temperature structural design methods and criteria used to design for such loadings are outlined. It is shown that the limiting design consideration for 2 1/4 Cr-1 Mo steel is likely to be the accumulation of creep-rupture damage during the relaxation of residual stresses arising from thermal transient loadings. It is concluded that although many data needs for a new alloy are readily apparent, others are unique to each particular material and are often identified only as the material is

employed in actual design.(author).

- 4158 DEVELOPMENT OF A 9 CR STEEL WITH IMPROVED STRENGTH AND TOUGHNESS. Bodine, G.C.; Chakravarti, B.; Harkness, S.D.; Owens, C.M.; Roberts, B.; Vandergriff, D.; Ward, C.T. (Combustion Engineering, Inc., Chattanooga, TN (USA)). pp 160-164 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

A development program was initiated to provide an alternate code-approved advanced ferritic alloy for LMFBR use. Approximately 30 laboratory heats were made of the type 9Cr-1Mo-Nb-V-W. In elevated-temperature properties the experimental alloys are superior to code-approved and candidate commercial steels. Improvements realized in weldability, toughness, and elevated-temperature strength and stability are discussed.

- 4159 TOUGHNESS OF 2 1/4CR-1MO STEEL FOR LMFBR STEAM GENERATOR APPLICATIONS. Copeland, J.F.; Yuen, J.L.; Offer, H.P. (General Electric Co., San Jose, CA (USA)). pp 151-154 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

The toughness of 2 1/4Cr-1Mo steel has been evaluated with regard to Liquid Metal Fast Breeder Reactor (LMFBR) steam generator tubesheet, shell, and nozzle region applications. Drop weight nil-ductility transition (NDT), Charpy V-notch (CVN), tensile, and thick compact tension specimen J-Integral tests were performed. J-Integral and CVN tests from thick section plates and weldments were performed to temperatures as high as 510°C. Toughness in the vicinity of room temperature is adequate, in these cases, but elevated temperature toughness is reduced by low energy tear fracture.

- 4160 MAIN CHARACTERISTICS OF 9% CHROMIUM-2% MOLYBDENUM-NIOBIUM-VANADIUM STEEL TUBES. INFLUENCE OF VARIATIONS OF COMPOSITION AND THERMAL TREATMENT. Gunz, G.; Pellicani, F. (Centre d'Etudes Vallourec, 59 - Aulnoye (France)); Houis, J.; Duquaire, B. (Usinor Service Metallurgique, Thionville (France)). pp 164-170 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

EM 12 steel is well established as a material for use in superheater tubes of conventional power plant. Furthermore, this steel exhibits an acceptable combination of hot tensile and creep strength, together with resistance to stress-corrosion cracking and liquid sodium decarburization, which renders it suitable for LMFBR steam generator applications. The mechanical properties of EM12 as a function of composition and heat treatment are described. Emphasis is placed on the influence of delta-ferrite on the impact and creep properties and the optimization of these

properties by control of the content of ferrite-forming elements and austenitizing temperature.

- 4161 TNO RESEARCH ACTIVITIES ON FERRITIC STEELS FOR LMFB STEAM GENERATORS. Schinkel, J.W. (Nijverheidsorganisatie TNO, Apeldoorn (Netherlands)). pp 82-86 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

The niobium stabilized 2 1/4Cr1Mo-steel showed satisfactory regarding hot cracking, cold cracking and reheat embrittlement. The sensitivity to hot and cold cracking is very low; some tendency for reheat embrittlement could be related to carbon level. Other chromium molybdenum steels showed low to moderate hot cracking susceptibility. The unstabilized 2 1/4Cr1Mo showed decarburization after 2000 h exposure at 520°C; 9Cr1Mo steel showed some chromium depletion at 550°C in sodium.

- 4162 BREEDER REACTOR PROGRAM IN THE USA. PAST, PRESENT AND FUTURE. Brewer, S. (Department of Energy, Washington, DC (USA)). Gensan Nenji Taikai Gijiroku; No. 11, 118-124, 154-155(1978). (In Japanese).

In the United States, commercial fuel reprocessing and demonstration test of plutonium breeder reactors were now postponed. LMFB project and schedule of FFTF and afterwards await the results of INFCE. However, this is not discarding the development of LMFBs. With the existing energy resources, the United States can have the opportunity to make careful and thorough study. Emphasis is placed on the research and development on new alternative types of fuel. FFTF going to be operated soon should provide effective means for the developments of FBR fuel and materials. High priority is to be retained for the test and development of sodium system hardware. The nuclear proliferation problem is not related to heat transfer and secondary systems; it is associated with the selection of fuel and fuel cycle. The whole program is centered around LMFB design and development. The target output will be 600×10^3 -- 700×10^3 MW. In the United States, now is the time to develop excellent products and to study the nuclear proliferation problem more carefully.

- 4163 FAST BREEDER REACTOR DEVELOPMENT PROGRAM IN FRANCE. OUTLOOK FOR PRACTICAL UTILIZATION. Rozenholz, M. (Novatome Industries, 92 - Le Plessis-Robinson (France)). Gensan Nenji Taikai Gijiroku; No. 11, 103-113, 138-147(1978). (In Japanese).

In France, the experimental fast reactor Rapsodie and the prototype fast reactor Phenix are now in operation. The commercial demonstration fast reactor Super-Phenix Mark I will attain the criticality in 1982, and start normal operation in 1983. Industrial setup for constructing fast reactors was made from 1976 to 1977, and NOVATOME was organized by (EA) and reactor component makers. EDF plans to materialize two power plants with fast breeder reactors in 1981 and total capacity of 8,000 -- 10,000 MWe by 1992. Following on the Mark I, the first plant may have the capacity of 1,200 MWe or more as optimal from the experiences of design and construction of the demonstration plant. Along with the reactor plants, CEA and COGEMA started preliminary design studies on fuel fabrication and reprocessing plants. In

this way, in the former half of 1980s, fast reactor power plants will be introduced in series.

- 4164 INTERACTION SOLID SOLUTION HARDENING IN 2.25CR-1MO STEEL. Klueh, R.L. (Oak Ridge National Lab., TN). Mater. Sci. Eng.; 35: 239-253(1978).

The creep behavior of annealed 2.25Cr-1Mo steel - a low-alloy ferritic steel - is complicated by the metallurgical processes that occur in the steel. These processes, which involve both solid solution and precipitation effects, give rise to creep curves that differ from a classical three-stage creep curve in that two steady-state stages occur. During the first steady-state stage, the creep rate is controlled by the motion of dislocations that contain atmospheres of carbon and molybdenum atom clusters, a process termed interaction solid solution hardening. Eventually, the precipitation of Mo_2C removes molybdenum and carbon from solution, and the creep rate increases to a new steady state where the creep rate is controlled by atmosphere-free dislocations moving through a precipitate field. These nonclassical curves occur at intermediate stresses. As the stress decreases, the first steady-state stage disappears because the dislocation velocity decreases and the molybdenum-carbon atmosphere will be able to diffuse with the dislocation. At high stresses a classical curve occurs when the creep rate is controlled by a combination of processes that operate in the two steady-state stages of the nonclassical curves.

- 4165 DATABANK FOR THE PROTOTYPE FAST REACTOR. Montgomery, K.R. (UKAEA). pp 250-256 of Generalized data management systems and scientific information. Paris, France; OECD Nuclear Energy Agency (1978).

This databank is a large set of files controlled by a management suite of COBOL data handling programs. The data, data dictionary and data directory are stored on exchangeable disc packs in the Random Access Index Sequential mode on an ICL 4/72 computer. It is a databank for the UKAEA Prototype Fast Reactor which stores data pertaining to the entire composition of the in-core components, concentrating particularly upon the fissile fuelled driver assemblies and the many experimental assemblies of both fuel pin and cladding/structural materials. At this time, the reactor is not yet operating at full power. This should be achieved in 1977 and the databank will then contain reactor operating history and reactor physics data, and also post-irradiation examination data for the discharged units. The databank will then represent a complete data compilation of in-reactor performance of fully documented materials. The current size of the databank is of the order of 100 megabytes and is expected to increase to approximately 200 megabytes over the next 2 years. The ICL 4/72 computer will be phased out during 1977 and replaced by an ICL 2980 computer. The Databank Management System adopted by ICL is IDMS (Integrated Database Management System), and it will be available on the 2980 computer. Initially, the databank will be moved on to the 2980 in its current form, but IDMS potential will be probed by moving selected sub-sets of the databank into this management system.

- 4166 DRYOUT CHARACTERISTICS AT LOW MASS VELOCITIES IN A VERTICAL STRAIGHT TUBE OF A STEAM GENERATOR. Roko, K.; Takitani, K.; Yoshizaki, A.; Shiraha, M. pp 429-434 of

- International heat transfer conference. Vol. 1. Toronto, Ontario; National Research Council of Canada (1978).
From 6. heat transfer conference; Toronto, Canada (7 Aug 1978).
Dryout characteristics of forced convective water flow in a vertical straight tube are experimentally studied in order to evaluate the practical performance and design of a LMFBF (Liquid Metal Fast Breeder Reactor) steam generator. Experiments were carried out over the range of mass velocity 160-550 kg/m²s and steam pressure 18-153 kg/cm², employing two different methods of heating such as sodium-heating and Joule-heating. First, a new correlation is proposed for the dryout quality in terms of steam pressure and mass velocity. Second, the characteristics of the wall temperature around the dryout point, such as the temperature gradient along tube axis and the oscillatory behavior, are studied in relation to the steam pressure and mass velocity.
- 4167 STUDIES OF WATERSIDE CORROSION AND HEAT TRANSFER ON LMFBF EVAPORATOR TUBES. Ralph, J.C.; Tomlinson, L.; Silver, P.J.B.; Lilley, W. pp 379-384 of International heat transfer conference. Vol. 4. Toronto, Ontario; National Research Council of Canada (1978).
From 6. heat transfer conference; Toronto, Canada (7 Aug 1978).
A useful method for studying heat transfer and waterside corrosion on short lengths of LMFBF evaporator tube, under representative conditions (350°C, 176 bar), is described. The technique allows simple control of the applied heat flux during steady running and of the temperature ramps during dryout transients. Waterside heat transfer coefficients based on outside wall temperature measurements are in general agreement with predictions for single phase and boiling heat transfer. The rate of waterside corrosion was found to be insensitive to changes in heat flux or steam quality.
- 4168 EXPERIMENTAL DETERMINATION OF CLADDING TEMPERATURE FIELDS IN THE CRITICAL REGIONS OF ROD BUNDLES WITH TURBULENT SODIUM FLOW AND COMPARISON WITH CALCULATIONS. Moeller, R.; Tschoeke, H. pp 29-34 of International heat transfer conference. Vol. 5. Toronto, Ontario; National Research Council of Canada (1978).
From 6. heat transfer conference; Toronto, Canada (7 Aug 1978).
Being the first safety barrier, the rod claddings of reactor core elements constitute a highly loaded and hence extremely important structural element. Their reliable and optimum design determines the operating safety and the economy of the reactor system. This calls among others for the precise knowledge of the three-dimensional temperature fields under all operation conditions conceivable. Such temperature fields were determined experimentally with an LMFBF 19-rod bundle model in turbulent sodium flow by means of a measurement technique especially suitable for high temperature liquid metal cooling. The most important results of the investigations are reported and compared with other experiments, global and local calculations on indicating the boundary conditions to be taken into account. The adaption of both computation methods to the experiments presented and suitable coupling of them are necessary. The significant geometry parameter in the duct wall zone remains to be investigated in further experiments.
- 4169 SODIUM NATURAL CONVECTION FROM A VERTICAL PLATE. Sheriff, N.; Davies, N.W. pp 131-136 of International heat transfer conference. Vol. 5. Toronto, Ontario; National Research Council of Canada (1978).
From 6. heat transfer conference; Toronto, Canada (7 Aug 1978).
There are many components in LMFBF's which are in contact with quiescent sodium, and consequently natural convective heat transfer conditions are of interest to the designers. This paper describes a sodium natural convection experiment in which the primary objective was to establish the validity of analytical correlations for a low Prandtl number fluid (Pr similar to 0.007). The experiment consists of an electrically heated vertical plate immersed in a sodium pool; the latter is cooled externally by a highly rated air cooler. Varying thermal conditions can be applied at the heater surface from six independently adjustable heater sections which comprise the heater. Constant heat flux conditions were used in all of the tests reported.
- 4170 STUDY OF TRANSIENT, BUOYANCY-INDUCED FLOW STRATIFICATION RELEVANT TO THE PRIMARY POOL OF A LIQUID-METAL-COOLED FAST BREEDER REACTOR. Jones, P. pp 137-142 of International heat transfer conference. Vol. 5. Toronto, Ontario; National Research Council of Canada (1978).
From 6. heat transfer conference; Toronto, Canada (7 Aug 1978).
The paper presents the results of a study of the transient motion of a vertical jet issuing into a cylindrical vessel following a sudden increase in the density of the jet. This vessel was an idealized, axisymmetric model of the reactor, where the sudden cooling of the primary sodium following a reactor trip was simulated by switching the flow from pure water to brine. The aim of the work was to investigate the criterion for buoyancy influence on the transient flow. Observation of the flow pattern was achieved either by flow visualization or, for more quantitative results, by introducing the brine at a different temperature from that of the pure water and monitoring the transient temperature field with an array of fast response thermocouples. On the basis of this work, it is concluded that the buoyancy influence can be described by an overall Richardson number and that typical reactor post-trip flows will be buoyancy influenced.
- 4171 FRAGMENTATION OF UO₂ SUB 2% AND UC DROPS ENTERING A SODIUM POOL. Ganguli, A.; Bankoff, S.G. pp 149-154 of International heat transfer conference. Vol. 5. Toronto, Ontario; National Research Council of Canada (1978).
From 6. heat transfer conference; Toronto, Canada (7 Aug 1978).
In this paper attention is focused on whether coarse premixing of fuel and coolant is possible according to the jet penetration hypothesis, with applications to a hypothetical accident in a liquid metal cooled nuclear reactor. For UO₂ SUB 2% or UC fuel drops entering a sodium pool, the rapid bubble growth and collapse phenomenon, followed by crust breakup due to jet penetration, is analyzed as a function of the system pressure and coolant temperature. It appears that the UO₂ SUB 2%-Na system fragments rapidly upon contact, for coolant temperatures above 300°C, while the corresponding temperature for the carbide fuel is around 600°C, at atmospheric pressures. For UO₂ SUB 2% but not for UC, there is a cut off pressure in the 0.1-3 MPa range,

above which the crust does not break up for a particular coolant temperature between 400 and 800°C. Higher sodium temperatures aid the break up for both fuels. One concludes that coarse premixing for UO₂, but not necessarily for UC, is difficult to attain.

- 4172 NEUTRON THEORY, BREEDING PROPERTIES AND DOUBLING TIME IN A BREEDER REACTOR. Mathieu, Ph.; Degeling, M. (Liege Univ. (Belgium)). Univ. Liege, Fac. Sci. Appl., Coll. Publ.; No. 74, 61-86(1978). (In French).

In this work the mathematical expression of the fundamental parameters of a breeder reactor, i.e. the breeding ratio BR and the doubling time TD, is established. The influences of various quantities entering the expression of these parameters and the way leading to the decrease of Tsub(D), are investigated. The decrease of Tsub(D) is classically obtained by an increase of the breeding ratio, of the core specific power and of the fuel burn-up and by a decrease of the plutonium inventory and of its residence time out of pile and of the losses in the reprocessing. For a given enrichment, the total breeding ratio is itself improved by an increase of the density and of the volumetric fraction of the fuel in the core and of the fertile material in the sheath, by a decrease of the structural material and of the number of control rods and by an increase of the thickness of the sheath. One has pointed out the influence of the flattening of the power distribution curve on the decrease of Tsub(D) and the role played in that way by the design of a heterogeneous core, in which the fertile material is judiciously arranged in small blocks in the fuel zone whose enrichment is uniform in opposite to cores with several enrichment zones. Finally a classical neutronic theory allows for the establishment of relationship between the critical size and the core composition by a one or several neutron groups model for bare cores and cores with reflectors in steady conditions.

- 4173 TRANSIENT RADIATIVE HEAT TRANSFER BETWEEN TWO EMITTING-ABSORBING MEDIA. Chan, S.H.; Cho, D.H. pp 367-372 of International heat transfer conference. Vol. 3. Toronto, Ontario; National Research Council of Canada (1978).

From 6. heat transfer conference; Toronto, Canada (7 Aug 1978).

Numerical (exact) solutions have been obtained for the transient energy exchange by radiation between two emitting-absorbing media that are suddenly brought into contact. Results for the temperature profile and radiative heat flux are presented for two different values of the initial temperature ratio (0.8 and 0.5). Also discussed are two approximate methods, one of which yields an analytical expression for the temperature profile valid for small times. An application of the solutions to accident analysis of liquid-metal-cooled fast breeder nuclear reactors is illustrated.

- 4174 HEAT TRANSFER IN REACTOR SAFETY. Tong, L.S. pp 285-309 of International heat transfer conference. Vol. 6. Toronto, Ontario; National Research Council of Canada (1978).

From 6. heat transfer conference; Toronto, Canada (7 Aug 1978).

This paper describes the basic heat transfer phenomena and the predictive techniques which are particularly important in nuclear reactor safety, and also presents the state-of-the-art of reactor heat transfer. Previous review

papers are identified. Emphasis is placed on recently published information, and areas are noted where research efforts should be encouraged.

- 4175 POWER INTO THE 21ST CENTURY. Foster, A.R. pp 151-181 of Convention of the Pan American Federation of Engineering Societies. Vol. 4. Santiago, Chile; Pan American Federation of Engineering Societies (1978).

From 15. Pan American Federation of Engineering Societies convention; Santiago, Chile (1 Oct 1978).

The sources of power which may be available in the future decades are considered. Solar energy, ocean thermal power, and fusion power systems are briefly reviewed, and it is predicted that coal and nuclear systems will have to fulfill the major part of the demands for energy. For systems using high sulfur coal, it is hoped that it will be possible to utilize a fluidized bed. On the other hand, if the sulfur content is low, sometimes there may be a problem with high sodium in the ash. Undoubtedly coal gasification will also be developed. A detailed solution is given to three nuclear systems, namely, the Canadian pressurized heavy water reactor (CANDU), the high temperature gas-cooled reactor using a direct gas turbine cycle (HTGR), and the breeder reactor (LMFBR), its controversy and its promise.

- 4176 TRIBOLOGICAL BEHAVIOUR OF MATERIALS FOR USE IN SODIUM COOLED FAST BREEDER STEAM GENERATOR APPLICATIONS. Hartle, R.T. (General Electric Co., San Jose, CA (USA). Breeder Reactor Dept.); Gallo, T.A.; Wilson, W.L. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). pp 320-325 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

Friction and self-welding behaviour of 2 1/4Cr-1Mo Vacuum Arc Remelted steel has been characterized by pin/plate tested to obtain dynamic and breakaway coefficients of friction. These tests were followed by steam generator tube/support plate model tests to obtain tube wear characteristics, under prototypical conditions (temperature, loading, and sodium environment). The average dynamic coefficients of friction ranged from 0.5 to 0.75 under contact loads from 88.5 N to 295.8 N. Breakaway friction factors ranged up to 1.50. Ingot Iron and Inconel 718 have been tested and evaluated as alternate support plate materials. Marked reduction in wear resulted with the use of Inconel 718 at both 502C and 446C.

- 4177 FERRITIC STEELS IN REACTOR TRIBOLOGY. Wild, E.; Mack, K.J. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). pp 326-330 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

The materials so far used as structural and fuel element cladding materials for sodium cooled fast reactors have been austenitic steels. However, their swelling behaviour under irradiation has negative consequences with

respect to stresses generated in the core. For this reason, ferritic steels are being discussed as a back-up solution. The tribological behaviour of two such grades of steel on a hard alloy and like on like has been investigated and the results have been compared with those obtained with austenitic steels. Their suitability for use in full-scale components was determined on the basis of test rigs in model scale geometries.

4178 CREEP BEHAVIOUR OF 2 1/4Cr1MoNb STEEL.

Husslage, W.; Dortland, W. (Nijverheidsorganisatie TNO, Apeldoorn (Netherlands)). pp 228-233 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London, England; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

The creep behaviour of Nb-stabilized 2 1/4Cr1Mo-steel was investigated by means of creep strain tests performed between 500 and 650°C up to 15,000 h on 30 mm plate material and on a welded joint in two stress relieved conditions. Results were compared with the rather meagre data in the literature. The strength under LMFBR design conditions was estimated by extrapolation with the Larson-Miller parameter. Special attention was paid to the creep retardation observed at high stresses and relatively low temperatures. A comparison was made with creep results obtained at 550°C 2 1/4Cr1MoNb in a ferritic condition.

4179 BILINEAR REPRESENTATION OF THE CYCLIC STRESS-STRAIN BEHAVIOR OF INCOLOY 800.

Maiya, P.S. (Argonne National Lab., IL). pp 1-13 of Inelastic behavior of pressure vessel and piping components. Chang, T.Y.; Krempl, E. (eds.). New York, NY; American Society of Mechanical Engineers (1978).

The bilinear stress-strain analysis for Incoloy 800 under cyclic-loading conditions is described. Although the method of determining the bilinear stress-strain parameters is based on a procedure proposed in the RDT Standard F9-1 for inelastic analysis of fast-flux-test-facility components, the accuracy and consistency of results have been improved by an analytical technique that includes certain simplifications. The bilinear stress-strain parameters of solution-annealed Incoloy 800 (heat HH7058A) under cyclic-loading conditions at a strain rate (ϵ /sub t/) of $8.6 \times 10^{-5} \text{ s}^{-1}$, total strain range ($\Delta\epsilon$ /sub t/) between 0.2 and 0.8%, and temperatures from 22 to 593°C have been determined. The dependence of bilinear cyclic-hardening parameters on strain ϵ /sub max/ (ϵ /sub max/ = $\Delta\epsilon$ /sub t/2) at various temperatures and their correlation with accumulated plastic strain are discussed.

4180 PROGRESS IN LIQUID-METAL FAST BREEDER REACTOR SAFETY RESEARCH AND DEVELOPMENT - AN OVERVIEW OF THE ISSUES.

Fontana, M.H. (Oak Ridge National Lab., TN (USA)). Prog. Nucl. Energy; 2: No. 3, 199-218(1978).

A comprehensive review is presented of the research and development directed towards an understanding of the events that could contribute to the release and transport of fission products and transuranic elements up to the boundary of the secondary containment of LMFBRs. The subject is discussed under headings corresponding to four fundamental lines of assurance which have been defined for the analysis of event sequences in this field

ie: accident prevention; limitation of core damage; containment of accident products in inner barriers (primary system); and attenuation of radiological products.

4181 ECONOMIC EVALUATION OF TOKAMAK POWER PLANTS.

Reid, R.L.; Steiner, D. (Oak Ridge National Lab., TN (USA)). pp 757-774 of Fusion reactor design concepts. Vienna; IAEA (1978).

From IAEA conference and workshop on fusion reactor design; Madison, WI, USA (12 Oct 1977).

In this study we determine the impact of plasma operating characteristics, engineering options, and technology on the capital cost trends of tokamak power plants. Tokamak power systems are compared to other advanced energy systems and found to be economically competitive. A three-phase strategy for demonstrating commercial feasibility of fusion power, based on a common-site multiple-unit concept, is presented.

4182 CORROSION OF 2 1/4 CR-1MO STEEL UNDER SUPERHEAT TRANSFER CONDITIONS.

Griess, J.C.; Devan, J.H. (Oak Ridge National Lab., TN (USA)); Hampton, L.V.; Challenger, K.D. (General Electric Corporation, USA). pp 367-270 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

Steam generators being developed for use with liquid metal fast breeder reactors (LMFBR's) will operate at substantially greater heat fluxes than conventional fossil-fired steam plants. To establish the corrosion allowance for the superheater tube walls, tests on 2 1/4Cr-1Mo steel with a heat flux of 126 kW/m² across the specimen surface were conducted under steam conditions similar to those that will exist in LMFBR superheaters.

4183 STRESS CORROSION STUDIES FOR 2 1/4Cr1Mo STEEL.

Indig, M.E. (General Electric Co., Pleasanton, CA (USA)). pp 408-412 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

Both the evaporators and the superheaters of future LMFBR steam generators are to be constructed with 2 1/4Cr-1Mo steel. One of the major concerns is the possibility that caustic contamination may occur during operation on the aqueous and steam sides of the steam generator. Under specific environmental and mechanical conditions caustic solutions are known to cause stress corrosion cracking of ferritic steels. For this reason the susceptibility of 2 1/4Cr-1Mo to caustic cracking was determined in badly faulted high temperature aqueous (316°C) and superheated steam environments (482°C). It was found that 2 1/4Cr-1Mo was extremely resistant to stress corrosion cracking in all metallurgical conditions tested over a wide range of stresses and oxidizing potentials. While no stress corrosion was detected at 316°C and 482°C, the rate of general corrosion was significant in the caustic environments. Major differences in corrosion rates were noted among different heats of the alloy at 316°C, while at 482°C all samples corroded excessively, 1.27 to $2.54 \times 10^{-3} \text{ mm/h}$ (0.05 to 0.1 mil/h). At lower

temperatures, 232°C corrosion rates were considerably reduced, and the alloy was readily cracked in constant extension rate tests. These studies form the basis for the operating of steam generator units and provide information for the handling of components under postulated off-chemistry conditions.

- 4184 CORROSION BEHAVIOUR OF IRON-CHROMIUM ALLOYS USED AS BOILER TUBE MATERIALS IN FAST REACTOR STEAM GENERATORS. Pritchard, A.M.; Knights, C.F.; Marsh, G.P.; Peakall, K.A.; Perkins, R.; Myatt, B.L.; Antill, J.E. (UKAEA, Harwell. Atomic Energy Research Establishment). pp 360-366 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

Experiments have been carried out, chiefly on 9Cr-1Mo material, using (i) capsules fabricated from tubing and containing various corrodents to study corrosion under static conditions, (ii) a high pressure water loop to study salt deposition under realistic mass flow and heat transfer conditions using radiotracers, (iii) a rotating disc assembly to study wastage mechanisms by corrosion and erosion in Na₂O-NaOH melts, and (iv) C-rings for constant strain and constant strain rate tests to assess susceptibility to cracking in concentrated NaOH solutions under controlled potential conditions.

- 4185 WEAR BEHAVIOUR OF Nb-STABILIZED 2 1/4Cr-1Mo STEEL TUBES IN SODIUM. Van Westenbrugge, J.K.; Beijer, N.H.H. (Neratoom N.V., The Hague (Netherlands)); Schinkel, J.W.; De Gee, A.W.J. (Nijverheidsorganisatie TNO, Apeldoorn (Netherlands)). pp 338-341 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

The inspection of the 50MW prototype straight tube steam generator tested in the 50MW-facility in Hengelo revealed some wear of the tubes. Corrective actions were taken for the SNR-300 straight tube steam generator design. To corroborate these actions tubes and supports have been tested in a specially designed tube wear test rig. In support small specimen testing has been performed. Both types of testing showed that the wear damage is small and will remain within acceptable limits.

- 4186 WELDING EXPERIENCE WITH TUBE-TO-TUBESHEET JOINTS OF STEAM GENERATORS IN 2 1/4Cr-1Mo STABILIZED FERRITIC STEEL. Beyer, N.H.H. (Neratoom N.V., The Hague (Netherlands)). pp 459-465 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

Tube bundles for the steam generators of the SNR 300 plant have been fabricated. In order to guarantee a high quality weld an extensive R and D program was successfully carried out. This program and the destructive and non-destructive testing methods used are described

in this paper.

- 4187 WELDABILITY AND JOINT PROPERTIES OF ADVANCED CARBON STABILIZED 2.25Cr-1Mo STEEL FOR LMFR. Terai, K.; Shimizu, S.; Nakai, Y.; Murakami, T. (Kawasaki Heavy Industries Ltd., Kobe (Japan)); Ota, S.; Fujiwara, M. (Kobe Steel Ltd. (Japan)). pp 442-446 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

A series of experimental tests have been carried out for weld cracking sensitivity and joint properties to establish the welding technique for 2.25Cr-1Mo-Nb and 2.25Cr-1Mo-Nb-Ni steels which have been newly developed as structural materials for LMFR steam generators. The results are as follows: (i) The weld cracking sensitivity of 2.25Cr-1Mo-Nb steel is nearly equivalent to that of 2.25Cr-1Mo steel; (ii) The creep rupture strength and fatigue strength of 2.25Cr-1Mo-Nb steel and 2.25Cr-1Mo-Nb-Ni steel are equal to, or superior to that of 2.25Cr-1Mo steel. During tests, all materials rupture within base metal, and the tensile strengths are satisfactory. The impact value of 2.25Cr-1Mo-Nb steel is relatively low; on the other hand, the impact value of 2.25Cr-1Mo-Nb-Ni steel is satisfactory.

- 4188 MECHANICAL PROPERTIES OF TRANSITION JOINT MATERIALS IN SUPPORT OF LMFR STEAM GENERATOR DESIGN. Brinkman, C.R.; King, J.F.; Strizak, J.P.; Klueh, R.L.; Booker, M.K. (Oak Ridge National Lab., TN (USA)). pp 496-503 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

Mechanical data needs are identified for transition joint weldments between austenitic and ferritic structural materials planned for LMFR service. Since the heat-affected zone in the ferritic material is apparently the critical area, particular attention must be given to behaviour in this region. Interim results are given to show differences in mechanical properties of the joint materials with particular emphasis on the ferritic materials.

- 4189 CRITICAL TECHNICAL ISSUES: LARGE LMFR PLANT. Anderson, C.A. Jr.; Fillnow, R.H. (Westinghouse Electric Corp., Madison, PA). Proc. Am. Power Conf.; 40: 255-265 (1978). (CONF-780440--).

From American power conference; Chicago, IL, USA (Apr 1978).

Problems anticipated in the development of large LMFR type reactors are reviewed.

- 4190 LOW-TEMPERATURE POOL-TYPE LARGE BREEDER. Cage, J.F.; Davies, S.M.; Sim, R.G. (General Electric Co., Sunnyvale, CA). Proc. Am. Power Conf.; 40: 266-277 (1978). (CONF-780440--).

From American power conference; Chicago, IL, USA (Apr 1978).

A low-temperature saturated cycle pool-type LMFR suitable for utility application in the United States is presented. The configuration of the primary heat transport system and the

selection of steam cycle are described. Breeder generation costs and variable (fuel) costs are also discussed.

4191 STATUS OF PHENIX AND SUPER PHENIX.

Zaleski, P. (European Nuclear Society, Paris, France). Proc. Am. Power Conf.; 40: 278-284(1978). (CONF-780440--).

From American power conference; Chicago, IL, USA (Apr 1978).

The operating history of Phenix and the construction status of Super Phenix are reviewed. Problems with the intermediate heat exchanger in Phenix are discussed in some detail.

4192 CONSERVATIVE POOL CONCEPT FOR A LARGE BREEDER REACTOR.

McDonald, J.S. (Rockwell International, Canoga Park, CA); Storrs, C.L.; Gills, C. Proc. Am. Power Conf.; 40: 285-291(1978). (CONF-780440--).

From American power conference; Chicago, IL, USA (Apr 1978).

A conceptual design of a large pool-type LMFBR is presented.

4193 TRANSITION JOINT WELDING DEVELOPMENT FOR LMFBR STEAM GENERATORS.

King, J.F.; Slaughter, G.M. (Oak Ridge National Lab., TN (USA)); Sullivan, M.D. (General Electric Co., San Jose, CA (USA)). pp 476-480 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

The objective of this study was to develop an improved transition weld joint between stainless steel and ferritic steel piping to circumvent the failures and concerns with joints made by existing technology. The primary concern of mismatch in coefficients of thermal expansion between the ferritic and austenitic steels and the high stresses it imposes at the interface between ferritic steel and weld metal led to an extensive elastic stress analysis involving base metal combinations, filler metals, and weld joint geometries. This analysis indicated that for pipe, stress imposed at the joint could be reduced considerably by using a transition material with an intermediate coefficient of thermal expansion between the 2 1/4Cr-1Mo ferritic steel and the austenitic stainless steel. Alloy 800H was selected as an appropriate intermediate material for this application. The advantage of using this material and the welding development involved in making these joints is described.

4194 WATERSIDE CORROSION OF STABILISED 2 1/4%Cr-1%Mo FERRITIC STEEL UNDER HIGH HEAT FLUX CONDITIONS.

Tomlinson, L.; Hurdus, M.H.; Silver, P.J.B.; Lilley, W. (UKAEA, Harwell. Atomic Energy Research Establishment). pp 353-359 of Ferritic steels for fast reactor steam generators. Pugh, S.F.; Little, E.A. (UKAEA, Harwell. Atomic Energy Research Establishment) (eds.). London; British Nuclear Energy Society (1978).

From British Nuclear Energy Society conference on ferritic steels for fast reactor steam generators; London, UK (30 May 1977).

The waterside corrosion of stabilised 2 1/4%Cr 1%Mo ferritic steel tubes has been studied under conditions which simulate the inlet region of a Prototype Fast Reactor (PFR) evaporator. The amount of corrosion occurring

in times from 1000 to 3400 hours was found to be unaffected by: thermal and pressure changes which simulate extreme conditions in the plant; 1 ppm NaCl or 7 ppm NaOH in the loop water; steam quality; heat flux; tube surface condition; the presence of large defects. Large defects in the heat flux zone were able to concentrate soluble species present in the loop water. The presence of a high heat flux (660 kW/m²) resulted in an increased deposition of non-porous magnetite onto the tube surface.

4195 SURVEILLANCE OF THE VIBRATIONAL BEHAVIOUR OF PRIMARY-CIRCUIT COMPONENTS.

Brillon, A.; Puyal, C. (Electricite de France, 78 - Chatou. Direction des Etudes et Recherches); Carre, J.C.; Tigeot, Y. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)). pp 377-393 of Nuclear power plant control and instrumentation, 1978. Vol. 1. Vienna, Austria; International Atomic Energy Agency (1978). (In French)

From IAEA symposium on nuclear power plant control and instrumentation; Cannes, France (24 Apr 1978).

Surveillance of the vibration of nuclear power station components can be performed by analysing fluctuations in the signals produced by ionization chambers and detecting elements. The paper contains a description of surveillance techniques applied to French fast neutron reactors (Phenix) and pressurized water reactors (Chooz, Fessenheim, Bugey). The detection of stray objects is also considered.

4196 SYSTEMS FOR DETECTING FAILED FUEL IN FAST REACTORS.

Berlin, C.; Jarretou, H.; Rulleau, A. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France)); Coulon, P. (CEA Centre de Marcoule, 30 - Bagnols-sur-CEze (France). Service de la Centrale Phenix); Pazlaud, A. (Novatome Industries, 92 - Le Plessis-Robinson (France)). pp 425-445 of Nuclear power plant control and instrumentation, 1978. Vol. 1. Vienna, Austria; International Atomic Energy Agency (1978). (In French)

From IAEA symposium on nuclear power plant control and instrumentation; Cannes, France (24 Apr 1978).

The failed-fuel detection systems for Rapsodie, Phenix and Super-Phenix are described in terms of their respective differences and design evolution. The technological problems encountered during design studies and construction are discussed. After considering in-service operating problems with failed-fuel detection and localization systems, the paper presents an account of results obtained in the detection of failed fuel.

4197 GASEOUS FISSION-PRODUCT DETECTOR FOR LMFBR FAILED-FUEL DETECTION.

Miyazawa, T.; Ashibe, K.; Takamatsu, S. (Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan). Research and Development Center); Mimoto, Y. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). pp 351-364 of Nuclear power plant control and instrumentation, 1978. Vol. 1. Vienna, Austria; International Atomic Energy Agency (1978).

From IAEA symposium on nuclear power plant control and instrumentation; Cannes, France (24 Apr 1978).

A new gaseous FP (fission products) detection apparatus has been developed for an LMFBR (liquid-metal fast-breeder reactor) failed-fuel detection system. Major components are three cylindrical chambers with fixed central wires, valves, control circuits with alarm units and radiation measuring circuits.

Each of the chambers is designed to have two functions - precipitation by electrostatic force to daughter nuclides of gaseous FP, and detection of the daughter nuclides by changing only the polarity of applied potential. The adoption of chambers can avoid problems with moving-wire type precipitators. In the development work, the apparatus was trial-fabricated and tested with a small sodium in-pile loop. Major performances obtained by the tests were a collection efficiency of 65 $\pm$ 15%; a detection efficiency of 73 $\pm$ 3%; major collected nuclides on wire surfaces were ^{86}Rb , ^{87}Rb and ^{137}Cs ; at about 100 V precipitation voltage, the collection efficiency was roughly saturated; the collection efficiency dependence on sampling gas flow-rate and on soaking time proved to agree well with the expected tendency at design stage; an accumulation of long-lived daughter nuclides was not found in the experimental period; and each component worked well, and few improvements have been indicated. In conclusion, it is quite possible, with a few improvements, for the developed apparatus to be practicable for a failed-fuel detection system.

- 4198 COMMISSIONING OF PROTOTYPE FAST-REACTOR CONTROLS. Collins, G.B. (UKAEA, Winfrith. Atomic Energy Establishment); Caney, V. (UKAEA, Risley, UK); Humphreys, P.D. (UKAEA, Dounreay. Nuclear Power Development Establishment); Thomsson, R.K. (Nuclear Power Co. (Risley) Ltd. (UK)). pp 109-125 of Nuclear power plant control and instrumentation, 1978. Vol. 1. Vienna, Austria; International Atomic Energy Agency (1978).

From IAEA symposium on nuclear power plant control and instrumentation; Cannes, France (24 Apr 1978).

The UKAEA Prototype Fast Reactor (PFR) and its control systems are described. System commissioning procedures were devised; their extent and use are examined in the context of several essential steam generator plant control loops. Feedwater regulation at low power is satisfactory, but problems have arisen during automatic change-over from 7% to 100% duty valves. At higher powers the regulator performance is only adequate for small or slow power perturbations. The problems of coping adequately with a rapid load manoeuvre, such as a turbine trip, are examined. Related problems with the turbine-driven feed pump were also observed, and these are shown to be mainly due to deficiencies in the electro-hydraulic servo-amplifier which controls the turbine steam admission valves. Finally, the requirements and design of an automatic controlled power setback system are described. This is designed to minimize shock to various plant components owing to rapid temperature changes when the turbo-alternator is tripped. Extensive use is made of dynamic models of the PFR, not only for control system design, but equally for commissioning and performance assessment.

- 4199 WELDING THE CORE SUPPORT PLATE FOR THE CLINCH RIVER BREEDER REACTOR PLANT, USA. Canary, W.W.; Franco-Ferreira, E.A. (Allis-Chalmers Corp., York, Pa. (USA)). pp 373-378 of Advances in welding processes. Cambridge, England; Welding Institute (1978).

From 4. national conference on advances in welding processes; Harrogate, UK (9 May 1978).

The core support structure for the USA's Clinch River breeder reactor plant consists of a 51mm thick cylindrical barrel welded to a 610mm thick core support plate, both in Type 304 stainless steel. The reactor plant will operate at temperatures in excess of 427°C and hence the design of the core support structure is not covered by the ASME code. One of the

more difficult weldments in the support structure was associated with the fabrication of the 610mm thick core support plate. It was made from two forgings comprising a 3.5m diameter forged disc placed inside a 4.5m diameter forged ring which were joined by a circular, 530mm thick, full penetration weld. A narrow gap submerged-arc welding process was successfully developed for this application.

- 4200 STUDY OF MECHANICAL CHARACTERISTICS OF STRUCTURAL MATERIALS FOR THE SNR PLANT. Scheibe, W.; Schmitt, R.; Di Primio, J.C.; Anderko, K. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). KFK Nachr.; 10: No. 3/4, 37-40(1978). (In German).

The present paper deals with studies on the austenitic steel X6CrNi 1811 in the unirradiated and the irradiated state, problems like monotonous hardening behavior, cyclic hardening and fatigue behavior as well as combined creep and fatigue stress being investigated.

- 4201 TRIBOLOGICAL STUDIES IN SODIUM. Wild, E.; Mack, K.J. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorbauelemente). KFK Nachr.; 10: No. 3: 4, 43-45(1978). (In German).

After the determination of pairs of materials with the highest resistance to wear and the best antifriction properties parameter studies are performed varying the following quantities: speed of causing friction, pressure per unit of area, geometry of specimens, sodium temperature, surface corrosion, and kind of motion.

- 4202 GAMMA HEATING MEASUREMENTS IN LMFBR SHIELDS WITH THERMOLUMINESCENT DOSIMETERS. Yoon, W.Y. Knoxville, TN; Univ. of Tennessee (1978). 211p. University Microfilms Order No. 79-03,486.

Thesis (Ph. D.).

An important and relatively new application for the thermoluminescent dosimeter (TLD) is the direct and simultaneous measurement of gamma-ray energy deposition distributions in heavy metal shields typical to fast breeder reactors. This dissertation describes the development of a TLD dosimetry system capable of measuring gamma-ray energy deposition in a mixed (neutron and gamma-ray) radiation field to within $\pm 3\%$ on an absolute basis. This work includes a detailed evaluation of the experimental techniques normally employed in TLD dosimetry, the verification of the gamma-ray spectral correction factor used to relate the energy deposition in the TLD to that occurring in the sleeve material, and the development of new neutron dose correction factors for both $\text{CaF}_2:\text{Mn}$ and ^7LiF thermoluminescent materials.

- 4203 FEATURES OF OPERATING CALCULATIONS OF PERMISSIBLE POWER FOR CHANNEL REACTORS AT BELOYARSKAYA ATOMIC ELECTRIC POWER PLANT. Sheinkman, A.G.; Petrov, V.A.; Kozyrev, V.D. (Izv Akad Nauk, USSR). Power Eng. (Engl. Transl.); 16: No. 6, 12-17(1978).

The classification is given and basic problems indicated for operational thermophysical calculations of permissible power for the channel reactors at the Beloyarskaya Atomic Power Plant (BAPP). The method is described and a flowchart given for a program to compute the permissible power of a BAPP reactor. The computational method used at the BAPP and the regulations for reactor-

parameter monitoring provide an adequate margin below the critical heat-transfer region in normal operation.

- 4204 (RISLEY-Trans--4010) BN-350 REACTOR. DEVELOPMENT, CONSTRUCTION AND OPERATING EXPERIENCE. Troyanov, M.F. Translated from Kernenergie; 21: No. 5, 775-787(1978). 15p. British Library Lending Division, Boston Spa, Wetherby, West Yorkshire, England.

The author analyses the experience gained with the BR-1, BR-2, BR-5 and BOR-60 reactors, which has served as the basis for the development of the BN-350 reactor with a thermal power of 1,000 MW and operating on ^{235}U . The most important design solutions for this reactor are discussed, together with the problems that arose during commissioning. Operating experience since then is described.

- 4205 MATHEMATICAL MODEL FOR AN INTEGRATED NUCLEAR POWER SYSTEM. Pavelescu, M. (Institutul de Reactori Nucleari Energetici, Pitesti (Romania)). Stud. Cercet. Fiz.; 30: No. 9, 899-928(1978). (In Romanian).

A mathematical model associated to an integrated nuclear power system constituted by two sub-systems, namely: a first one represented by PHWRs and LMFBRs, and a second one represented by PWRs and the same LMFBRs is presented. It is assumed that both sub-systems are developing in a first stage only in thermal variants, and in a second stage in an integrated (mixed) thermal-fast variant. The development of a fast reactor is allowed by the amounts of Pu reprocessed in the nuclear system. The independent variables of the model are the development periods of the sub-systems and the time variable, while the dependent variables are the produced year energy amounts and year energy specific cost prices. The paper is illustrated by numerous numerical results obtained by application of the integrated nuclear power system mathematical model, which allow a lot of different conclusions.

- 4206 OUTLOOK OF THE FAST BREEDER REACTOR DEVELOPMENT IN U.S.S.R. Krasnolarov, N. (Nauchno-Issledovatel'skij Inst. Atomnykh Reaktorov, Dimitrovgrad (USSR)). Gensan Nenji Taikai Gijiroku; No. 11, 113-118, 148-153(1978). (In Japanese).

From 11. annual conference of the Japan Atomic Industrial Forum; Tokyo, Japan (14 - 16 Mar 1978).

In USSR, following on the experimental fast reactor BOR-60, the large fast reactor BN-350 started operation in 1973. Its electric output is 120 MW, and its desalination capacity is 70,000 m³ per day. This is of loop type. On the other hand, a pool type fast reactor, BN-600, is now under construction and completion of the main part is expected in 1979. With the operation of this type, therefore, USSR will have experiences of both loop and pool types. These two should lead to future commercial power generation. The next step in the FBR development program is BN-1600. It will have the largest electric output, using two turbines of 800 MW each. It is the pool type, and it has the specification parameters similar to those of BN-600.

- 4207 PROJECT OF DEVELOPING FAST BREEDER REACTORS IN JAPAN. Oyama, A. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Gensan Nenji Taikai Gijiroku; No. 11, 124-128, 156-159(1978). (In Japanese).

From 11. annual conference of the Japan Atomic Industrial Forum; Tokyo, Japan (14 -

16 Mar 1978).

The government decided "The policy concerning the development of power reactors" in 1968, and 10 years have elapsed since the development of fast breeder reactors has become one of the national projects in Japan. Since the oil crisis in 1973, the expectation to nuclear power generation grew larger and larger in view of the stable securing of energy in Japan. Recently, various discussions on the utilization of plutonium and the development of fast breeder reactors are made on the basis of the nuclear non-proliferation policy of USA, but breeder reactors are indispensable for Japan, which depends on import for around 90% of energy resources, therefore largest effort to put the breeder reactors into practical use must be exerted. The personnel of the PNC engaging in this project is 550 and the budget is 24 billion yen in 1978. The fast experimental reactor "Joyo" attained the criticality in April, 1977, and the measurement and test reactor performance were carried out in low power output operation. Now the reactor is stopped for the inspection. The design of the fast breeder prototype reactor "Monju" is almost finished, and the construction will begin in near future. The reasons for having selected the type and power output of the prototype "Monju" are explained. The international cooperation with USA, UK and FRG and the road to the practical use hereafter are described.

- 4208 Les surregenerateurs. (BREEDERS). Palaiseau, France; Societe Francaise d'Editions et d'Informations Regionales (1978). 72p. (In French).

The interest lying in the nuclear energy as a source of electric power is recalled with a view to the consumption of primary energy and electric power; the position of France is particularly discussed. The fast breeder reactor is presented and the state of development of this type of reactor in France is discussed with the planning of the Creys-Malville power plant. The problems successively examined are concerned with: fuel cycle, radioactivity and effluents, breeder safety, licensing procedures, sodium coolant, plutonium, fuel reprocessing, environmental impact and waste management.

- 4209 DEVELOPMENT WORK ON CORE PROTECTIVE INSTRUMENTATION. Jacobi, S.; Rohrbacher, H.A.; Schleisiek, K. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung); Edelmann, M.; Ehrhardt, J. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik); Heckert, K. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung); Krebs, L.; Mueller, S. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorbauelemente); Voges, U. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Datenverarbeitung in der Technik); Stieglitz, L. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Heisse Chemie). KFK Nachr.; 10: No. 3-4, 109-115(1978). (In German).

For the SNR 300 there is provided, according to the present design stage, a detection method by means of delayed neutrons (DND system) and the measurement of the fuel outlet temperature for monitoring the fuel subassemblies. These two methods therefore also form the keypoint of the development work for the nuclear instrumentation that is reported in this paper.

- 4210 INFLUENCE OF THE SODIUM COOLANT ON THE MATERIALS CHARACTERISTICS. Borgstedt, H.U.; Frees, G. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). KFK Nachr.; 10: No. 3-4, 41-42(1978). (In German).

In the present paper the following problems are discussed: effect of sodium compared to the effects of other coolants; sodium influence in the coolant loop; phenomena of sodium corrosion; and effects on the mechanical properties of reactor components.

- 4211 THERMOHYDRAULICS OF THE CORE OF FAST NEUTRON REACTORS. Khairallah, A.; Lapparent, D. de; Jallade, M. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Service d'Etude et de Developpement des Coeurs); Mouney, H.; Martin, J. (Electricite de France, 75 - Paris. Direction des Etudes et Recherches). Entropie; No. 83, 4-10(1978). (In French).

This article describes the main problems involved in cooling the core of fast-neutron reactors. The applications concern SUPER PHENIX 1 or PHENIX. After a brief description of SUPER PHENIX 1, the following problems are dealt with: the thermohydraulics of the fuel pencil bundles, taking into consideration deformations of the assemblies by exposure to radiations; the thermohydraulics of the core, i.e. the distribution of the flow throughout the core and the control of the flow per assembly; the estimations of the maximum temperature of the cladding; the thermohydraulics of the immediate environment of the core, i.e. the grid upstream and the space between the core and the core cover downstream. Finally, special operating characteristics are mentioned, such as the failure of normal power outlet circuits and boiling due to a cooling fault.

- 4212 SODIUM-WATER REACTIONS IN THE STEAM GENERATORS OF FAST NEUTRON NUCLEAR POWER STATIONS. Lions, N. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides); Quinet, J.-L. (Electricite de France, 75 - Paris. Direction des Etudes et Recherches). Entropie; No. 83, 41-49(1978). (In French).

This very great chemical affinity of sodium for water could have important consequences for the availability of fast-neutron nuclear power stations, if there were a leak in the heat exchange wall of the steam generators. In this article, after a brief description of the different types of leak, the chemical reaction is described and then the calculation codes developed by the EDP and the CEA to determine the effects on the secondary circuit of a sodium-water reaction. The results of these theoretical and experimental studies are being used by the principal manufacturer (NOVATOME-NIRA) of the SUPER PHENIX boiler to ensure that the plant is safe.

- 4213 RECENT KFK WORK ON IRRADIATION TECHNIQUES FOR FAST REACTORS. Bojarsky, E.; Haefner, H.E.; Mueller, K.; Schmidt, L. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). KFK Nachr.; 10: No. 3-4, 27-28(1978). (In German).

In the not too distant future there may be used with KNK-2 the first German power reactor with a fast core for irradiation experiments. The test zone, offering a central fuel element position and six rim positions, is of special interest because of its relatively high fast

neutron flux. But also in the driver zone respectively near the reflector there are planned irradiation experiments. The central position being directly accessible from above through the closure head, there may be performed here preferably such experiments requiring supplies of pipes and cables to the test insert for instrumentation, control etc., as e.g. fuel element boiling experiments, accident experiments as well as investigations on cladding and structural material.

- 4214 ANALYSIS OF SEVERE HYPOTHETICAL ACCIDENTS. Froehlich, R.; Maschek, W.; Schmuck, P. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik); Royl, P.; Struwe, D. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). KFK Nachr.; 10: No. 3-4, 66-71(1978). (In German).

A loss-of-coolant-flow accident for the Kalkar nuclear power plant is not energetic, i.e. there will result a negligible mechanical stress on the reactor coolant system. In addition there can be shown that the masses of core material, which will temporarily melt, can be permanently cooled within the reactor tank. This rather satisfactory confirmation of the expected accident development is a main result of the R and D program of the last years. In order to be able to qualify this progress it has to be considered that such hypothetical accidents are very complex phenomena, the prediction of which has certainly been improved but which are still subject to some uncertainties. For the Kalkar nuclear power plant, in order to cover these uncertainties, there were simulated so-called energetic limiting cases besides the expected accident development. Also for these limiting cases there will result a radiation exposure for the environment not exceeding the limits of the Radiation Protection Ordinance.

- 4215 FACILITY FOR NUCLEAR AEROSOL EXPERIMENTS. Alexas, A.; Hoffmann, G.; Jordan, S.; Kind, J.; Lindner, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Lab. fuer Aerosolphysik und Filtertechnik 1). KFK Nachr.; 10: No. 3-4, 119-120(1978). (In German).

FAUNA is a plant that was designed for large-scale investigations on sodium spray and areal fires, on aerosol behavior in the closed containment, and on the development of aerosol separators, and that was taken into operation at the end of 1978. In the present paper there is described the plant with its safety systems and its instrumentations. Moreover the experimental program of FAUNA is presented.

- 4216 EXPERIMENTAL AND THEORETICAL STUDIES ON THE PRESSURE BUILD-UP OF CORE MATERIALS IN HYPOTHETICAL ACCIDENTS. Bober, M.; Fischer, E.A.; Karow, H.U.; Thurnay, K. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). KFK Nachr.; 10: No. 3-4, 92-96(1978). (In German).

The present paper is treating the following classes of problems: theoretical extrapolations of the equations of state for pure oxide fuel; experimental determination of the vapor pressure diagram of pure oxide fuel up to 5000 K; and contribution of the fission products to the pressure build-up in the fuel, thermodynamic variables of state for sodium.

- 4217 RESULTS OF THE IN-PILE EXPERIMENTAL

PROGRAMM SCARABEE. Bottoni, M.; Struwe, D. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung); Billiaux, M. (Societe Belge pour l'Industrie Nucleaire, Brussels); Eibel, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung); Jacobs, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). KFK Nachr.; 10: No. 3-4, 75-78(1978). (In German).

Description of the in-pile program SCARABEE having been performed in the CABRI reactor is presented. The aim of it was to investigate the effects of cooling disturbances on the fuel rod behavior including failure. The special features of the test facility and the results of the theoretical interpretation of the transient thermal-hydraulic behavior are presented.

- 4218 STUDIES ON AEROSOL BEHAVIOR UNDER ACCIDENT CONDITIONS.** Bunz, H.; Jordan, S.; Schikarski, W.; Schuetz, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Lab. fuer Aerosolphysik und Filtertechnik I). KFK Nachr.; 10: No. 3-4, 116-118(1978). (In German).

Studies are presented on aerosol formation by sodium evaporation and condensation, on activity release from the bottom cooling unit, on aerosol formation by sodium fires and on the behavior of gasborne aerosols in the containment.

- 4219 THEORETICAL INVESTIGATIONS OF FUEL-COOLANT INTERACTION.** Caldarola, L.; Ladisch, R. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung); Jacobs, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). KFK Nachr.; 10: No. 3-4, 79-80(1978). (In German).

In the present paper there are discussed the reaction mechanisms of the system Na-UD_2 and the influence of the temperature on the development of the reactions. Many experiments indicate that a temperature criterion must be fulfilled for an explosive interaction to take place.

- 4220 EXPERIMENTS ON FUEL-COOLANT INTERACTION UNDER TRANSIENT CONDITIONS.** Deckers, H.; Lehning, H.; Piel, D.; Reiser, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung); Gross, S. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Projekt Schneller Brueter). KFK Nachr.; 10: No. 3-4, 81-82(1978). (In German).

In no one of the experiments carried out an energetic reaction with shock waves and a high energy conversion factor was measured, as it may occur e.g. between different molten metals and water. Though there were produced pressure pulses with amplitudes up to about 70 bar their width and number is too small to cause serious loads on a reactor structure. The same is true for the conversion of thermal energy of the fuel into mechanical energy of the coolant.

- 4221 EXPERIMENTAL STRESS ANALYSIS IN FAST NUCLEAR REACTOR SAFETY.** Curioni, S.; Trombetti, T. pp 483-485 of 7th congress on materials testing. Vol. 2. Budapest, Hungary; OMKDK-Techninform (1978).
From 7. congress on materials testing;

Budapest, Hungary (9 Oct 1978).

In the primary containment design of a liquid metal fast breeder reactor (LMFBR) the possibility of a sudden energy release in the core (hypothetical core disruptive accident) must be considered. This paper discusses some simple vessel tests which have been carried out to verify the general pattern of structural response. 7 refs.

- 4222 SAFETY SYSTEMS RELATIVE TO THE CREYS-MALVILLE POWER STATION.** Noel, H.; Freslon, H. (Novatome Industries, 92 - Le Plessis-Robinson (France)); Lucenet, G. (Societe Centrale Nucleaire Europeenne a Neutrons Rapides (NERSA), 69 - Lyon (France)). Bull. Inf. Sci. Tech. (Paris); No. 227, 81-98(1978). (In French).

At Creys Malville, as in all nuclear power stations, public protection against any accidental discharge of radioactive products and its potential consequences is based on the setting up in series of barriers of specified tightness. The safety of the station therefore rests on the effectiveness of each of these barriers and its correct behaviour under the different working conditions of the station. Part one will analyse the various barriers in turn distinguishing for each on the three basic progressive stages of safety: prevention, supervision and the attendant safety action. The analysis of the different working conditions of the station given in part two will enable role and behaviour of each barrier for these various conditions to be represented as a whole.

- 4223 THERMOHYDRAULICS OF STEAM GENERATORS.** Ginier, R. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides); Robin, M.G. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Service d'Etude et de Developpement de la Technologie des Reacteurs a Sodium); Gonnet, B. (Societe Franco-Americaine de Constructions Atomiques (FRAMATOME), 92 - Courbevoie (France)); de Montardy, A. (Electricite de France, 75 - Paris). Entropie; No. 83, 33-40(1978). (In French).

This article deals with the thermohydraulics of steam generators in pressurized-water or fast-neutron reactors built in France. A list is given of the theoretical or experimental methods used to analyse either the behaviour of the steam generator as a whole or phenomena occurring in more localised areas of the exchange surface. Details are given of the amount of precision of the methods used, and the fields where research is still needed are pointed out. The methods used have made an extensive study of these appliances possible, but research on certain point is still necessary: knowledge of two-phase flow and its models must be improved, in particular to get away from one-dimensional schematization. Great efforts must be developed to the study of complying between flow structures, phenomena of mass transfer at the wall, and the corrosion which this may lead to.

- 4224 POWER FLATTENING BY ENRICHMENT ZONING IN A FAST POWER REACTOR.** Lee, S.M.; John, T.M.; Singh, R.S. (Reactor Research Centre, Kalpakkam (India)). pp 392-395 of Reactor physics. Bombay, India; Department of Atomic Energy (1978).

From Symposium on reactor physics; Bombay, India (1 Mar 1976).

The use of two enrichment zones for radial power flattening in a Pu-U oxide fuelled large

LMFBR has been studied. Besides the choice of zone enrichments, optimization of the relative volumes of the two core zones has been made from the point of view of minimum radial form factor, maximum power per unit fissile mass and minimum flux gradients. It was found that an inner zone to outer zone volume ratio of 55/45 is optimum with a corresponding zone enrichments ratio of 0.68 for a 3000 litre core.

- 4225 NEUTRONIC CONSIDERATIONS IN THE DESIGN OF A FUEL PIN FOR A FAST POWER REACTOR. Lee, S.M.; John, T.M.; Singh, R.S. (Reactor Research Centre, Kalpakkam (India)). pp 396-399 of Reactor physics. Bombay, India; Department of Atomic Energy (1978).

From Symposium on reactor physics; Bombay, India (1 Mar 1976).

Results of neutronic calculations, performed in connection with the design of a helium bonded mixed plutonium-uranium oxide fuel pin for a large LMFBR, are reported. Some of the design features of such a pin affecting the core neutronic performance are the pellet diameter, the fuel smear density, the clad thickness, the active pin length and the axial blanket configuration. Optimisation of the pin diameter on the basis of the doubling time, assuming a reactor thermal power of 1500MW, average linear pin power of 290 W/cm and using currently available sodium pumping powers gave an optimum pellet diameter in the range 7.5 mm to 8 mm. The relative changes in doubling time due to variations in the smear density and the clad thickness have been also calculated. A study of the penalty paid in fissile inventory on flattening the core due to sodium pumping power limitations indicated a recommended minimum H/D ratio of 0.4 thus fixing the active pin length to be around 100 cms. The effect on the axial breeding contribution due to the use of integrated or separate blanket pins was estimated and resulted in the recommendation of having at least the bottom blanket pins integrated.

- 4226 CRACK PROPAGATION INVESTIGATION RELATED TO THE LEAK-BEFORE-BREAK CONCEPT FOR LMFBR PIPING. Begley, J.A. (Ohio State Univ., Columbus (USA)); Wilson, W.K.; Sheinker, A.A. pp 215-224 of Tolerance of flaws in pressurized components. London, England; Mechanical Engineering Pub. Ltd. for the Institution of Mechanical Engineers (1978).

From Conference on tolerance of flaws in pressurized components; London, UK (16 May 1978).

Results are presented from a study to evaluate leak-before-break conditions in LMFBR (Liquid Metal Fast Breeder Reactor) piping systems. Emphasis is placed on piping elbows, but most conclusions are also applicable to straight cylindrical sections. Critical crack sizes in pressurized elbows are experimentally measured and shown to agree well with predictions made using a plastic instability model. Also, surface crack fatigue growth patterns are determined. This information is combined to demonstrate a leak-before-break rationale.

- 4227 ULTRASONIC FLOW-METER TEST IN SODIUM. Ishii, Y.; Uno, O.; Kamei, M. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). pp 439-453 of Nuclear power plant control and instrumentation 1978. Vol. 2. Vienna, Austria; International Atomic Energy Agency (1978).

From IAEA symposium on nuclear power plant

control and instrumentation; Cannes, France (24 Apr 1978).

As a part of the R and D programme for the prototype fast breeder reactor MONJU, an ultrasonic flow-meter (USFM) test is being carried out in sodium in the O-Arai Engineering Center of PNC. Prior to the present test, an in-water test was done at the manufacturer's as a preliminary investigation. The results reported here are the results up to the present. Calibration tests using the actual fluid were conducted on a 12-inch ultrasonic flow-meter with guide rods fabricated for sodium flow measurement.

- 4228 LOW-AMPLITUDE STEADY STATES IN A SODIUM HEATED STEAM GENERATOR. Llory, M. (Electricite de France, 78 - Chatou. Direction des Etudes et Recherches). pp 337-352 of Phenomenes thermiques et hydrauliques nonstationnaires. Paris, France; Eyrolles (1978). (In French)

Indicial analysis tests of the Super Phenix 45 MW nominal power experimental steam generator were carried out in 1973 on the sodium heated steam generator test system of Electricite de France at the Renardieres. The numerical study of the dynamic behavior of this apparatus was made by means of a non-linear thermo-hydraulic calculation programme developed by EDF for simulating thermal or nuclear units (SICLE code). After touching on the simulation method and the main calculation hypotheses associated therewith, the calculation results achieved are commented, comparing them with the experimental results. The peculiarities of the modelisation of steam generators with helically wound tubes are also dealt with, as are the correlations adopted for determining the friction and thermal transmission coefficients.

- 4229 EFFECTS OF RADIATION ON STRUCTURAL MATERIALS: PROCEEDINGS OF THE INTERNATIONAL SYMPOSIUM, 9TH, 1978. Sprague, J.A.; Kramer, G. (eds.). (US Nav Res Lab, Washington, DC). Am. Soc. Test. Mater., Spec. Tech. Publ.; No. 683, 1-684(1978).

From 9. international symposium on the effects of radiation on structural materials; Richland, Washington, USA (11 Jul 1978).

This symposium contains 42 papers, all of which are indexed separately. The papers cover the broad area of radiation damage of materials. They are divided into six subject areas: microstructure, fracture of thermal reactor materials, primary damage production and irradiation creep simulation, dosimetry and damage function analysis, mechanical properties, and creep.

- 4230 HIGH-INTEGRITY THERMOCOUPLES FOR IN-PILE APPLICATIONS. Thomson, A.; Evans, R.A. (UKAEA Risley Nuclear Power Development Labs.). pp 401-419 of Nuclear power plant control and instrumentation, 1978. Vol. 2. Vienna, Austria; International Atomic Energy Agency (1978).

From IAEA symposium on nuclear power plant control and instrumentation; Cannes, France (24 Apr 1978).

Temperature measurements in the coolant of an LMFBR are usually made with metal-sheathed, mineral insulant thermocouples. When considerations of speed of response or available space require the cable to be immersed in the coolant, or its diameter to be small, the desired reliability may not be attained with conventional thermocouples. It is shown how a coaxial design can reduce the risk of failure caused by sodium penetration of the

sheath or metallurgical deterioration of the conductor. A 1.0 mm diameter co-axial thermocouple is described, which has a thermal response time of about 50 ms, a much higher reliability than the conventional mineral-insulated (MI) thermocouple and is incapable of sustaining an undisclosed operational fault.

- 4231 LOW-CYCLE FATIGUE OF THREE IRRADIATED AND UNIRRADIATED VANSTAR ALLOYS. Korth, G.E.; Schmunk, R.E. (Edgerton, Germeshausen and Grier Idaho, Inc, Idaho Falls). Am. Soc. Test. Mater., Spec. Tech. Publ.; No. 683, 466-476(1978).

From 9. international symposium on the effects of radiation on structural materials; Richland, WA, USA (11 Jul 1978).

Results are reported for strain-controlled low-cycle fatigue measurements on Vanstar-7, 8, and 9 in both the irradiated and unirradiated conditions, conducted at 400°C. Specimens were irradiated in the experimental breeder reactor II (EBR-II) to fluences of 0.18 to 6.37*10²⁵ neutrons (n)/m² (E>0.1 MeV) at 410 to 450°C. The results indicate that neutron irradiation had little or no effect on the fatigue life of any of the alloys. Comparison with data for Types 304 and 316 austenitic stainless steels tested at 400°C shows that on the basis of strain range the irradiated Vanstar alloys are inferior or equivalent to the stainless steels below 10,000 cycles-to-failure but become superior above that point. Proportional limit and yield-strength information have been extracted from the fatigue data, where possible. 8 refs.

- 4232 IN-REACTOR DEFORMATION AND FRACTURE OF AUSTENITIC STAINLESS STEELS. Bloom, E.E.; Wolfer, W.G. (Oak Ridge Natl Lab, Tenn). Am. Soc. Test. Mater., Spec. Tech. Publ.; No. 683, 656-672(1978).

From 9. international symposium on the effects of radiation on structural materials; Richland, WA, USA (11 Jul 1978).

An experimental technique for determining in-reactor fracture strain was developed and demonstrated. Differential swelling between a specimen holder and a test specimen with a lower swelling rate produced uniaxial deformation in 304 and cold-worked 316 stainless steel specimens. In-reactor deformations of 0.7 to 2.1 percent were achieved in Type 304 stainless steel previously irradiated to fluences up to 8.8x10²⁴ neutrons (n)/m² without fracture. These strains are significantly higher than found in postirradiation creep-rupture tests on similar specimens. From the measured strain values and published irradiation creep data and correlations, the stress levels during the irradiation were calculated. 9 refs.

- 4233 CONTROL SYSTEM ADOPTED FOR SUPER-PHENIX. REASONS FOR CHOICE AND EVALUATION OF PERFORMANCE. Decuyper, J.; Skull, G. (Societe Centrale Nucleaire Europeenne a Neutrons Rapides (NERSA), 69 - Lyon (France)); Hery, M.; Hennebicq, J.P. (Electricite de France, 75 - Paris); Acket, C. (Novatome Industries, 92 - Le Plessis-Robinson (France)). pp 129-141 of Nuclear power plant control and instrumentation, 1978. Vol. 2. Vienna, Austria; International Atomic Energy Agency (1978). (In French)

From IAEA symposium on nuclear power plant control and instrumentation; Cannes, France (24 Apr 1978).

The paper reviews all the research done in working out the control system for the fast-neutron Super-Phenix power station, which is

now under construction at Creys-Malville, France. The purpose of the system is to provide a balance between the power produced by the reactor and that taken by the electricity-generating plant. After an introductory section on the structure of the power station and the operating conditions imposed, the following main stages in design work are described: development of the system simulation model and corroboration on the basis of test results; specification of possible control system layouts (i.e. the various possible connections between regulating variables and regulated variables), optimization of control coefficients of each layout, comparison of performance and choice of layout; detailed study of the layout chosen. Special reference is made to the following typical aspects of Super-Phenix operating technology: response of the power station to primary frequency control; stability of steam generators operating in parallel; establishment of the sodium temperature value. The final part is a summary of the research carried out and a description of the performance of the computer codes.

- 4234 EVOLUTION OF THE CORE SURVEILLANCE AND PROTECTION SYSTEM IN FAST REACTORS. Gourdon, J.; Barbier, G.; Peirano, F. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France)); Jeannot, A.; Le Bourhis, A. (Novatome Industries, 92 - Le Plessis-Robinson (France)). pp 23-40 of Nuclear power plant control and instrumentation, 1978. Vol. 2. Vienna, Austria; International Atomic Energy Agency (1978). (In French)

From IAEA symposium on nuclear power plant control and instrumentation; Cannes, France (24 Apr 1978).

The evolution of the core surveillance and protection system in French fast reactors from Rapsodie to Super-Phenix is described. The main trends in neutron flux measurements, core temperature surveillance, fuel failure detection and the control rod system are indicated and developments now under way are reviewed.

- 4235 DATA PROCESSING AND DATA COLLECTION IN SUPER-PHENIX. Josue, M. (Novatome Industries, 92 - Le Plessis-Robinson (France)); Thegner, G. (Sodeteg-TAI, Buc, France); Skull, G. (Societe Centrale Nucleaire Europeenne a Neutrons Rapides (NERSA), 69 - Lyon (France)). pp 143-158 of Nuclear power plant control and instrumentation, 1978. Vol. 2. Vienna, Austria; International Atomic Energy Agency (1978). (In French)

From IAEA symposium on nuclear power plant control and instrumentation; Cannes, France (24 Apr 1978).

The data processing systems for the Super-Phenix power station have been developed from Phenix systems, the various tasks being specified on the basis of the origin of information (specific to the boiler or common to the whole power station) and of its nature, i.e. depending on whether it is used for protection or for operational purposes or whether it provides personnel with a better understanding of phenomena related to the reactor. The data processing systems specific to the boiler are as follows: (a) the core temperature processing system (TRTC) with which fuel assembly temperatures can be monitored and any abnormally high value discovered, in which case it can cause a trip to shut down the reactor. To this extent it can be seen as part of the station safety equipment. In the interest of channel separation and satisfactory availability, the system is made up of two

identical units based on the use of mini-computers, some of which (for analog acquisition) are decentralized and placed near the measuring points in the dome; and (b) the core fault detection and diagnosis system (DDDC), which is a necessary complement to the TRTC in that it fulfils certain boiler operation tasks and supplies information if incidents occur. It is made up of three subsystems (acquisition and retrieval, reactivity comparison, noise analysis). Among the systems applicable to the overall operation of the station, there is: (c) the complementary information processing system (TCI) which provides overall control and is based on a large quantity of information connected with the facility as a whole.

- 4236 PRACTICAL POWER-REACTOR REACTIVITY METER. Peterson, B.R.; Larson, H.A. (Argonne National Lab., Idaho Falls, ID (USA)). pp 483-499 of Nuclear power plant control and instrumentation, 1978. Vol. 2. Vienna, Austria; International Atomic Energy Agency (1978).

From IAEA symposium on nuclear power plant control and instrumentation; Cannes, France (24 Apr 1978).

An on-line reactivity and anomalous reactivity meter has been designed and constructed at EBR-II. The meter is capable of identifying unknown reactivity perturbations associated with single subassembly events, sensing an unscheduled loss-of-pumping-power with sensitivity equal to the flow-meter, recording reactivity data during reactor experiments, and calibrating the control rods at start-of-run. The meter is based upon a minicomputer which senses power level via a fission and calculates reactivity and anomalous reactivity by using inverse kinetics. It operates over a wide range of power, from six dollars subcritical through 125% of full power.

- 4237 COLLECTION AND TRANSMISSION OF MILLI-VOLT-LEVEL DATA ON A LIQUID-METAL FAST-BREEDER REACTOR. Robinson, C. (Nuclear Power Co. (Risley) Ltd. (UK)); Thoburn, R. (Reyrolle Parsons Automation Ltd., Gateshead, Durham, United Kingdom). pp 65-79 of Nuclear power plant control and instrumentation, 1978. Vol. 2. Vienna, Austria; International Atomic Energy Agency (1978).

From IAEA symposium on nuclear power plant control and instrumentation; Cannes, France (24 Apr 1978).

The paper describes a high integrity data multiplexing system which caters for up to 252 input elements with remote measuring facilities, using a 12-core flexible cable. Tests carried out on a prototype unit and recommendations for effecting improvements are described. The system was developed following the identification of the problem of having to handle a large number of cables on the top of the Commercial Demonstration Fast Reactor. It has high reliability, a relatively small and conventional cable installation and is applicable to situations outside the nuclear field.

- 4238 PROTECTION OF THE SODIUM-COOLED FAST REACTOR DURING CORE MELT-DOWN. Efimov, I.A.; Lopatin, Yu.V.; Mameev, L.I.; Rysev, A.M.; Stabrovskij, S.A.; Ushakov, V.M.; Filonov, V.S. (Nauchno-Issledovatel'skij Inst. Priborostroeniya, Moscow (USSR)). pp 55-62 of Nuclear power plant control and instrumentation, 1978. Vol. 2. Vienna, Austria; International Atomic Energy Agency (1978). (In Russian)

From IAEA symposium on nuclear power plant control and instrumentation; Cannes, France (24 Apr 1978).

The paper describes a system based on delayed neutron detection which is designed to give protection against serious accidents involving core melt-down in a fast reactor. The system automatically normalizes delayed neutron data to reactor power with correction for neutron detection time. The alarm signal is generated if the normalized value of the readings rises above a threshold equivalent to the geometric area of the fuel in the fuel element and if the growth rate of the process is too rapid for operator analysis.

- 4239 REASSESSMENT OF THE ROLE OF STRESS IN DEVELOPMENT OF RADIATION-INDUCED MICROSTRUCTURE. Garner, F.A.; Wolfer, W.G.; Brager, H.R. (Westinghouse Hanford Co, Richland, Wash). Am. Soc. Test. Mater., Spec. Tech. Publ.; No. 683, 160-183(1978).

From 9. international symposium on the effects of radiation on structural materials; Richland, WA, USA (11 Jul 1978).

Data are now accumulating which clearly demonstrate that the stress state plays a strong role in the development of void and dislocation microstructure in metals during neutron irradiation. In these experiments the application of a tensile biaxial stress state at constant fluence and temperature has been found to lead to a progressively decreasing metal density with increasing stress. The effect of stress on the concurrent development of voids, Frank interstitial loops. Whereas a previous analysis of these data invoked as the dominant process an effect of stress on the rate of thermal re-emission of vacancies by voids, this process has been found to be too slow at the temperature at which the experiments were conducted to account for the magnitude of the observed perturbation. The current model employs as the dominant process the effect of stress on changing the capture efficiency of voids for interstitials. 28 refs.

- 4240 HELIUM PRODUCTION IN FAST BREEDER REACTOR OUT-OF-CORE STRUCTURAL COMPONENTS. Simons, R.L. (Westinghouse Hanford Co, Richland, Wash). Am. Soc. Test. Mater., Spec. Tech. Publ.; No. 683, 365-379(1978).

From 9. international symposium on the effects of radiation on structural materials; Richland, WA, USA (11 Jul 1978).

The major sources of helium production in stainless steel irradiated in a fast reactor are (n, α) reactions in nickel, iron, chromium, and impurities such as nitrogen and boron. The corresponding reaction cross sections are generally of the threshold type with threshold energies about 1 MeV. Two important exceptions are the cross sections for the two-stage nickel reaction and the boron-10 reaction. Although lifetime helium concentrations are low, they may be an important consideration, when applying material property change data obtained from the experimental breeder reactor II (EBR-II) core to high-temperature out-of-core spectra than for the harder in-core spectra in which most irradiation data are obtained. 11 refs.

- 4241 TENSILE PROPERTY CORRELATIONS FOR HIGHLY IRRADIATED 20 PERCENT COLD-WORKED TYPE 316 STAINLESS STEEL. Fish, R.L.; Cannon, N.S.; Wire, G.L. (Westinghouse Hanford Co, Richland, Wash). Am. Soc. Test. Mater., Spec. Tech. Publ.; No. 683, 450-465(1978).

From 9. international symposium on the effects of radiation on structural materials;

Richland, WA, USA (11 Jul 1978).

Recent experiments on developmental fast-flux test facility (FFTF) cladding (20 percent cold-worked Type 316 stainless steel) have extended the data base to a fast neutron fluence of 8.4×10^{22} neutrons (n)/cm² ($E > 0.1$ MeV). The specimens were irradiated in the experimental breeder reactor-II (EBR-II) at temperatures ranging from 371 to 816°C, although peak fluence levels were attained on specimens irradiated near 371 and 649°C only. Tension tests were performed at 232°C, near the irradiation temperature, and, in some cases, above the irradiation temperature. Test specimen strain rates ranged from 4×10^{-2} /s to 4×10^{-3} /s. The data generated on cladding irradiated near 371°C established that the low-temperature strength and ductility are fluence independent beyond about 5×10^{22} n/cm² ($E > 0.1$ MeV). The resulting correlations provide a description of strength and ductility over the temperature range of 371 to 871°C and strain rates of 10^{-6} to 10^1 /s. 11 refs.

- 4242 EFFECTS OF NEUTRON IRRADIATION AND FATIGUE ON DUCTILITY OF STAINLESS STEEL DIN-1-4948. de Vries, M.I.; van der Schaaf, B.; Staal, H.U.; Elen, J.D. (ECN, Petten, Neth.). Am. Soc. Test. Mater., Spec. Tech. Publ.; No. 683, 477-489(1978).

From 9. international symposium on the effects of radiation on structural materials; Richland, WA, USA (11 Jul 1978).

Test specimens of stainless steel DIN 1.4948, which is similar to AISI Type 304, have been irradiated at 723 and 823 K up to fluences of 1×10^{23} and 2×10^{23} neutrons (n)/cm² ($E > 0.1$ MeV). These are representative conditions for the reactor vessel and inner components of the liquid metal fast breeder reactor SNR-300 after 16 years of operation. High-temperature (723 to 1023 K) tension tests at strain rates ($\dot{\epsilon}$) from 10^{-5} to 10^{-1} /s show a considerable decrease of tensile ductility. The extent depends on helium content, test temperature, and strain rate. The atomic helium fractions of 3×10^{-4} and 7×10^{-4} result from the reactions of thermal neutrons with the 14 ppm boron present in the steel. Helium embrittlement sets in at strain rates below 1 to 10^{-4} /s (the range of interest for Bethe-Tait accident analyses). A minimum total elongation value of 6 percent is shown at 923 K. The postirradiation fatigue life is reduced by up to about 50 percent due to intergranular cracking. The combination of irradiation and fatigue causes a decrease of ductility after a smaller number of prior fatigue cycles than in the case of unirradiated material. 8 refs.

- 4243 ESTIMATES OF TIME-DEPENDENT FATIGUE BEHAVIOR OF TYPE 316 STAINLESS STEEL SUBJECT TO IRRADIATION DAMAGE IN FAST BREEDER AND FUSION POWER REACTOR SYSTEMS. Brinkman, C.R.; Liu, K.C.; Grossbeck, M.L. (Oak Ridge Natl Lab, Tenn.). Am. Soc. Test. Mater., Spec. Tech. Publ.; No. 683, 490-510(1978).

From 9. international symposium on the effects of radiation on structural materials; Richland, WA, USA (11 Jul 1978).

Cyclic lives obtained from strain-controlled fatigue tests at 593°C of specimens irradiated in the experimental breeder reactor II (EBR-II) to a fluence of 1 to 2.63×10^{26} neutrons (n)/cm² ($E > 0.1$ MeV) were compared with predictions based on the method of strain-range partitioning. It was demonstrated that, when appropriate tensile and creep-rupture ductilities were employed, reasonably good estimates of the influence of hold periods and irradiation damage on the fully reversed

fatigue life of Type 316 stainless steel could be made. After applicability of this method was demonstrated, ductility values for 20 percent cold-worked Type 316 stainless steel specimens irradiated in a mixed-spectrum fission reactor were used to estimate fusion reactor first-wall lifetime. The ductility values used were from irradiations that simulate the environment of the first wall of a fusion reactor. Neutron wall loadings ranging from 2 to 5 MW/m² were used. 27 refs.

- 4244 FRACTURE TOUGHNESS OF IRRADIATED AISI 304 AND 316L STAINLESS STEELS. Dufresne, J.; Henry, B.; Larsson, H. (CEA, Fontenay-Aux-Roses, Fr.). Am. Soc. Test. Mater., Spec. Tech. Publ.; No. 683, 511-528(1978).

From 9. international symposium on the effects of radiation on structural materials; Richland, WA, USA (11 Jul 1978).

The fracture toughness of AISI 304 and 316L stainless steels and their welds irradiated to 7×10^{25} neutrons (n)/m² (50 percent of this was at $E > 0.1$ MeV) was investigated on 10- and 20-mm-thick three-point bend specimens. Measurements of critical values of the J-integral, $J_{sub c}$, and the crack-opening displacement, $\Delta_{sub c}$, defined at the onset of stable crack growth, were performed at 23 and 400°C. The electric potential method was adapted to hot-cell operation for the detection of the initiation point. Drastic reductions of fracture toughness occur when combining welding, irradiation, and high temperature. As an example, the value $\Delta_{sub c}/0.51$ mm of base-material nonirradiated specimens tested at 20°C drops to 0.011 mm for welded irradiated specimens tested at 400°C. The consequences on liquid metal fast breeder reactor (LMFBR) structural design are illustrated in the case of the French LMFBR Super Phenix. 8 refs.

- 4245 EFFECTS OF IRRADIATION TEMPERATURE, FLUENCE, AND HEATING RATE ON POSTIRRADIATION FLOW PROPERTIES OF CLADDING UNDER SIMULATED TEMPERATURE TRANSIENT HEATING AND DEFORMATION CONDITIONS. Cannon, N.S.; Duncan, D.R. (Westinghouse Hanford Co, Richland, Wash.). Am. Soc. Test. Mater., Spec. Tech. Publ.; No. 683, 557-566(1978).

From 9. international symposium on the effects of radiation on structural materials; Richland, WA, USA (11 Jul 1978).

Irradiation temperature, fluence, and heating rate effects on plastic flow and failure properties of fast reactor cladding were investigated by heating internally pressurized specimens until failure occurred. Specimens tested were from 20 percent cold-worked Type 316 stainless steel cladding, irradiated in the experimental breeder reactor II (EBR-II) at temperatures to 720°C and fluences to 10^{23} neutrons (n)/cm² ($E > 0.1$ MeV). A decrease of transient failure temperature with increasing irradiation temperature was observed at the "standard" heating rate of 5.6 °C/s. This effect became more pronounced for irradiation temperatures above 600°C. Transient failure temperatures also generally decreased with increasing fluence for tests performed at the standard heating rate. 5 refs.

- 4246 EFFECTS OF IRRADIATION CREEP ON EX-REACTOR MECHANICAL PROPERTIES. Duncan, D.R. (Westinghouse Hanford Co, Richland, Wash.). Am. Soc. Test. Mater., Spec. Tech. Publ.; No. 683, 567-577(1978).

From 9. international symposium on the effects of radiation on structural materials; Richland, WA, USA (11 Jul 1978).

Irradiation creep of reactor core components

is usually considered nondamaging. This concept has had no experimental verification, reducing confidence in its application reactor design. The work reported herein was intended to test the damaging nature of irradiation creep. Material used was 20 percent cold-worked AISI 316 stainless steel tubing of 5.84-mm outside diameter and 5.08-mm inside diameter. The tubing specimens has been irradiated with stress and with zero stress to provide varying amounts of irradiation creep. A transient burst test was used to assess creep effects. This test imposes a constant stress and an increasing constant rate temperature ramp on a tubular specimen until rupture occurs. 12 refs.

- 4247 IRRADIATION CREEP IN BENDING OF COLD-WORKED AISI 316 STAINLESS STEEL AT LOW NEUTRON FLUENCE. McSherry, A.J.; Patel, M.R.; Marshall, J.; Appleby, W.K. (GE, Sunnyvale, Calif.). Am. Soc. Test. Mater., Spec. Tech. Publ.; No. 683, 598-607(1978).

From 9. international symposium on the effects of radiation on structural materials; Richland, WA, USA (11 Jul 1978).

The results from the first and second interim examinations of a test to measure irradiation creep in bending of 20 percent cold-worked AISI 316 stainless steel are presented. These low-fluence results indicate that irradiation creep in bending exhibits a larger primary creep component of the total strain as compared with creep in biaxial pressurized tubes of the same heat of material, but the secondary creep rates in the two cases appear to be similar. The data also indicate that the bending strains have a linear fluence and stress dependency, and strains measured on beams fabricated parallel to and transverse to the direction of cold work are similar, indicating that material texture anisotropy does not effect irradiation creep in bending. 6 refs.

- 4248 COMPARISON OF FUEL PIN DEFORMATIONS WITH PRESSURIZED TUBE CREEP TESTS. Taylor, J.P.; Gilbert, E.R.; Lovell, A.J. (Westinghouse Hanford Co, Richland, Wash.). Am. Soc. Test. Mater., Spec. Tech. Publ.; No. 683, 608-623(1978).

From 9. international symposium on the effects of radiation on structural materials; Richland, WA, USA (11 Jul 1978).

Fuel pins are a key component in the fast breeder reactor (FBR), and the economical performance of the reactor depends upon the behavior and life of fuel pin assemblies. One of the limiting phenomena in fuel pin and assembly performance is deformation. A thermomechanical performance code "SIFAIL" is being used at the Hanford Engineering Development Laboratory (HEDL) to predict fuel cladding deformation. SIFAIL utilizes empirical descriptions of cladding strain as a function of temperature, stress, and neutron fluence to predict fuel pin profiles. The equations are derived primarily from in-reactor tests on tubes pressurized with an inert gas and irradiated under conditions of constant temperature and stress. These equations were modified in SIFAIL to account for the variations in temperature and stress with time which occur in fuel pins. It is the purpose of these investigations to develop analytical methods which can be used to design fuel pins with optimum long-life performance. Equations were developed to describe the in-reactor creep behavior for two different classes of AISI 316 stainless steel tubing, a developmental class for the fast test reactor (FTR) and the FTR first-core tubing. The behavior of these two tubing classes was sufficiently different that

two distinctly separate equations were required to describe their in-reactor creep behaviors. 13 refs.

- 4249 PYROCHEMICAL PROCESSING OF LWR AND LMFBF FUELS BY THE SALT TRANSPORT METHOD. Knighton, J.B. (Rockwell International Corp., Golden, CO (USA). Rocky Flats Plant). Radiochim. Acta; 25: No. 3/4, 181-190(1978).

The salt transport pyrochemical process is being evaluated for coprocessing uranium and plutonium in spent LWR and LMFBF oxide fuel. The main objective is to develop a proliferation resistant process for recycle of uranium and plutonium. Diversion and proliferation resistance against weapons is enhanced by leaving part of the fission product elements with the uranium-plutonium product. The cladding (zircaloy or stainless) is removed from the oxide fuel by dissolution in molten zinc. The fuel oxides are then reduced by calcium metal in the presence of a Cu-Mg alloy and a $\text{CaCl}_2\text{-CaF}_2$ salt. The subsequent fission product separations from uranium and plutonium are achieved by a series of molten salt-metal equilibrations. Vacuum distillation is used to recover the volatile solvent metals for recycle. The uranium and plutonium product with residual fission products is subsequently converted to oxide fuel. The chemistry of the salt transport step is described. The integration of the salt transport step into a conceptual flow sheet is described.

- 4250 IMPACT OF PFR REPROCESSING AT DOUNREAY NUCLEAR DEVELOPMENT ESTABLISHMENT. Lillyman, E. (UKAEA Dounreay Nuclear Power Development Establishment). pp 117-126 of Radiation protection in nuclear power plants and the fuel cycle. Vol. 2. London; British Nuclear Energy Society (1978).

From Conference on radiation protection in nuclear power plants and the fuel cycle; Bristol, UK (27 Nov 1978).

This paper examines some of the consequences of changing over from an enriched uranium to a plutonium fuel cycle. It is concerned mainly with the irradiated fuel reprocessing plant which has been reconstructed to meet the standards required for a plutonium fuel cycle. The modifications have covered containment, shielding, ventilation, HEPA filtration and active posting systems.

- 4251 RADIOLOGICAL PROTECTION IN THE PROTOTYPE FAST REACTOR, DOUNREAY, SCOTLAND. Adam, E.R.; Howard, R.S.; McDonald, W. (UKAEA Dounreay Nuclear Power Development Establishment). pp 213-218 of Radiation protection in nuclear power plants and the fuel cycle. Vol. 1. London; British Nuclear Energy Society (1978).

From Conference on radiation protection in nuclear power plants and the fuel cycle; Bristol, UK (27 Nov 1978).

The Prototype Fast Reactor (PFR) at Dounreay is the largest power plant (250 MWe, 600 MW(th)) operated by the United Kingdom Atomic Energy Authority. The reactor was first taken critical in March 1974 and experience has since been obtained in the various aspects of radiological protection during almost 30,000 hours of operation at power levels up to and including the design thermal power of 600 MW. Radiation and contamination control has been highly satisfactory with personnel exposure levels extremely low and restrictions on personnel access applying to only a few limited areas. The paper gives a brief description of the reactor design shielding, describes the methods and facilities available for handling active components and discusses the

effectiveness of the radiation and contamination control policy being operated.

4252 NUCLEAR REACTOR WITH SEVERAL CORES.

Swars, H. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,628,782/A/. 29 Dec 1977. 11p. (In German).

Several sodium-cooled cores in separate vessels with removable closures are placed in a common reactor tank. Each individual vessel is protected against the consequences of an accident in the relevant core. Maintenance devices and inlet and outlet pipes for the coolant are also arranged within the reactor tank. The individual vessels are all enclosed by coolant in a way that in case of emergency cooling or refuelling each core can be continued to be cooled by means of the coolant loops of the other cores.

4253 IMPROVEMENTS IN OR RELATING TO THE BLOCKING OF MICROPORES IN A STRUCTURE.

Hailestead, W.D. (to Central Electricity Generating Board, London (UK)). British Patent 1,495,561/A/. 21 Dec 1977. 6p.

A method is described for blocking micropores that may occur during the manufacture of a sodium-water heat exchanger. Such pores may be so fine, say 10⁻²mm diameter or less, that they escape detection during testing. One face of the members to be treated is exposed to vapour phase Ni carbonyl, oxide, or hydroxide, whilst the other face is exposed to a suitable reactant that causes deposition of Ni or an oxide thereof in the micropores. If an oxide is deposited this is subsequently reduced to Ni by passing a reducing gas over the face that is to be exposed to Na.

4254 HEAT EXCHANGERS. Hayden, O.; Willby, C.R. (to UKAEA London Office). British Patent 1,495,654/A/. 21 Dec 1977. 4p.

Reference is made to "tube-in-shell" heat exchangers. Hitherto it has been the practice to seal the tubes to a tube sheet by welding, but frequently the welds fail, probably as a result of stress corrosion, and repair is a very difficult operation because there are so many closely spaced tubes. In the arrangement described welding is avoided by employing what are known as compression pipe couplings, the construction of which is described. The arrangement is particularly applicable to liquid metal cooled reactors.

4255 NUCLEAR REACTOR CORE SERVICING APPARATUS.

Andrea, C. (to Combustion Engineering, Inc.). US Patent 4,064,000. 20 Dec 1977. Filed date 29 Sep 1975. 26p.

Disclosed is an improved core servicing apparatus for a nuclear reactor of the type having a reactor vessel, a vessel head having a head penetration therethrough, a removable plug adapted to fit in the head penetration, and a core of the type having an array of elongated assemblies. The improved core servicing apparatus comprises a plurality of support columns suspended from the removable plug and extending downward toward the nuclear core, rigid support means carried by each of the support columns, and a plurality of servicing means for each of the support columns for servicing a plurality of assemblies. Each of the plurality of servicing means for each of the support columns is fixedly supported in a fixed array from the rigid support means. Means are provided for rotating the rigid support

means and servicing means between condensed and expanded positions. When in the condensed position, the rigid support means and servicing means lie completely within the coextensive boundaries of the plug, and when in the expanded position, some of the rigid support means and servicing means lie without the coextensive boundaries of the plug.

4256 RUPTURE DISC. Newton, R.G. (to Energy Research and Development Administration). US Patent 4,064,003. 20 Dec 1977. Filed date 29 Mar 1976. 6p. PAT-APPL-671,912.

The intermediate heat transport system for a sodium-cooled fast breeder reactor includes a device for rapidly draining the sodium therefrom should a sodium-water reaction occur within the system. This device includes a rupturable member in a drain line in the system and means for cutting a large opening therein and for positively removing the sheared-out portion from the opening cut in the rupturable member. According to the preferred embodiment of the invention the rupturable member includes a solid head seated in the end of the drain line having a rim extending peripherally therearound, the rim being clamped against the end of the drain line by a clamp ring having an interior shearing edge, the bottom of the rupturable member being convex and extending into the drain line. Means are provided to draw the rupturable member away from the drain line against the shearing edge to clear the drain line for outflow of sodium therethrough.

4257 IDENTIFICATION OF FAILED FUEL ELEMENT.

Fryer, R.M.; Matlock, R.G. (to Energy Research and Development Administration, Washington, DC (USA)). Canadian Patent 1,023,065/A/. 20 Dec 1977. 7p.

The invention relates to an improvement in the Experimental Breeder Reactor No. 2 (EBR-II) whereby existing fuel handling equipment can be used to interrogate each subassembly in turn to determine whether a failed fuel element is present therein.

4258 NUCLEAR CORE DEBRIS COLLECTING TRAY.

Broadley, D. (to United Kingdom Atomic Energy Authority). US Patent 4,062,724. 13 Dec 1977. Priority date 5 Apr 1974, United Kingdom of Great Britain and Northern Ireland (UK), 4p.

Disclosed is a nuclear reactor construction comprising a fast reactor core submerged in a pool of liquid sodium in a primary vessel. A collecting tray for core debris is submerged in the pool below the core. The collecting tray comprises a base plate constituting a plane tube sheet while the wall of the tray constitutes an annular tube sheet. Coolant conducting tubes are end received in the tube sheets. An internal skirt of the primary vessel overlaps the wall of the tray. 2 claims, 2 figures.

4259 NUCLEAR REACTORS. Jackson, G.O. (to United Kingdom Atomic Energy Authority). US Patent 4,061,534. 6 Dec 1977. Filed date 6 Oct 1971. 6p.

Disclosed is a nuclear reactor cooled by a freezable liquid has a vessel for containing said liquid and comprising a structure shaped as a container, and cooling means in the region of the surface of said structure for effecting freezing of said liquid coolant at and for a finite distance from said surface for providing a layer of frozen coolant on and supported by said surface for containing said liquid

coolant. In a specific example, where the reactor is sodium-cooled, the said structure is a metal-lined concrete vault, cooling is effected by closed cooling loops containing NaK, the loops extending over the lined surface of the concrete vault with outward and reverse pipe runs of each loop separated by thermal insulation, and air is flowed through cooling pipes embedded in the concrete behind the metal lining. 7 claims, 3 figures.

- 4260 METHOD FOR COOLING A BREEDER REACTOR AND BREEDER REACTOR FOR APPLYING THE METHOD. Gast, K. (to Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 1,940,014/C/. 1 Dec 1977. 4p. (In German).

The fuel assemblies of the fission zone and the breeder subassemblies in the radial breeding blanket are supported on a double bottom. The coolant gets into the interspace of the double bottom below the blanket. This part of the space is separated from the interspace of the double bottom below the fission zone. Each breeder subassembly consists of a twin tube. The coolant enters the inner tube, flows through it upwards in axial direction, is then deflected on the upper end, and afterwards, in the annulus of the twin tube, flows down again in axial direction into the inlet region, below the double bottom. From there on it flows upwards through the fuel assemblies of the fission zone from below. Thereby a uniformly high coolant outlet temperature is obtained.

- 4261 CREEP FAILURE TIME OF 20% COLD-WORKED TYPE 316 STAINLESS STEEL. Chakrapani, B. (Duke Power Co., Charlotte, N.C. (USA)). Nucl. Eng. Des.; 44: No. 3, 421-432 (Dec 1977).

The Robinson failure criterion is examined for its accuracy to predict the creep failure time of 20% cold-worked type 316 stainless steel under uniaxial and multiaxial stresses. Observed changes of slope in the log-log plots of maximum tensile stress versus isothermal rupture life are neglected, and large errors in prediction of creep failure time are found. A failure criterion that best explains the experimental behavior of 20% cold-worked type 316 stainless steel in uniaxial and multiaxial creep conditions is developed by incorporating the effective stress responsible for crack initiation and the maximum tensile stress responsible for crack propagation into the creep failure time equation.

- 4262 LARGE ELECTRO-MAGNETIC PUMP FOR HIGH TEMPERATURE LMFBR APPLICATIONS. Yang, C.C.; Kraus, S. (General Electric Co., Sunnyvale, Calif. (USA)). Fast Breeder Reactor Dept.). Nucl. Eng. Des.; 44: No. 3, 383-395 (Dec 1977).

A 14 500 gal/min (3300 m³/hr) electro-magnetic (EM) axial annular linear induction pump with center return (ALIP-CR) was designed to satisfy the requirements of Section III and Code Case 1592 of the ASME Boiler and Pressure Vessel Code. One of the important features of this EM pump is that there are no moving parts. Every significant loading condition was considered in the stress analysis of this pump for a liquid metal (Na) fast breeder reactor (LMFBR) application. An axisymmetric overall stress model (finite element) was first set up to analyze the stress state within the pump duct which has a pressure boundary with the conventional pressure vessel features. Axisymmetric loading such as pressure and thermal loads as well as non-axisymmetric loading such as seismic and vibration loads were taken into account. Local analyses were

then performed for specific critical areas to supplement the overall model and to substantiate the adequacy of the design. Analytical results show that the EM pump is an acceptable and attractive design for high temperature LMFBR applications.

- 4263 (ANL-Trans--1114) DEVICE FOR SUPPORTING THE CORE OF A FAST NEUTRON REACTOR. Lleres, J.; Martin, J.P.; Perona, M.; Venot, R. Dec 1977. Translation of French Patent No. 7,336,035 9p. Dep. NTIS, PC A02/MF A01.

The object of the invention is a supporting device for the core of a fast reactor cooled by a molten sodium circulation comprising a tank, a flooring mounted to this tank, a grid equipped with studs for the entry of the fuel elements, the grid which is supported by said flooring, a large number of tubes feeding the heat-transferring liquid sodium to the grid, characterized by the fact that said flooring consists of a rigid annular structure equipped with a plane annular bearing, that said grid includes a plane annular supporting bearing, seating of the grid on the flooring being assured by the support of said annular bearing of the grid on said annular supporting bearing of the flooring, and that the tubes opening into the grid to feed molten heat-transferring sodium are each connected to a structure connected to the flooring by means of an expansion bellows.

- 4264 (ANL-Trans--1113) INSTALLATION FOR PURIFYING THE LIQUID METAL USED FOR COOLING THE CORE OF A FAST NEUTRON REACTOR. Abramson, R.; Delisle, J.P.; Elie, X. Dec 1977. Translation of French Patent No. 7,335,377 12p. Dep. NTIS, PC A02/MF A01.

The invention deals with a purification installation for liquid metal and particularly sodium used to provide for cooling of the fuel assemblies of a fast-neutron reactor.

- 4265 (ANL-Trans--1112) TURNSTILE DEVICE FOR FUEL UNLOADING OF NUCLEAR REACTORS. Allain, A.; Mulot, P.; Filloleau, E. Dec 1977. Translation of French Patent No. 7,412,839 13p. Dep. NTIS, PC A02/MF A01.

The invention deals with a turnstile transfer device for handling the fuel assemblies of a nuclear reactor, in particular of the fast neutron type cooled by circulating molten metal, generally sodium.

- 4266 (ANL-Trans--1111) PROCESS FOR THE CONTINUOUS CONTROL OF THE SEAL AND STRENGTH OF THE CONTAINMENT OF A REACTOR UTILIZING SAID PROCESS. Cachera, P.C. Dec 1977. Translation of French Patent No. 7,332,374 7p. Dep. NTIS, PC A02/MF A01.

The object of the invention is a process for the continuous monitoring of the seal and mechanical strength of the containment of a sodium-cooled fast reactor and the application of said process.

- 4267 RADIATION RISKS FROM PLUTONIUM RECYCLE. Cuddihy, R.G. (Lovelace Biomedical and Environmental Research Inst., Albuquerque, NM); McClellan, R.O.; Hoover, M.D.; Dugan, V.L.; Chapman, L.D.; Wayland, J.R. Environ. Sci. Technol.; 11: No. 13, 1160-1164 (Dec 1977).

A computer simulation method was used to estimate the health risks associated with alpha-emitting radionuclides that could be released to the environment from a nuclear reactor program with plutonium recycle and from burning low-sulfur, Western U.S. coal. Many of the base

assumptions for the model were taken from the LMFBR Environmental Impact Statement (EIS). The probabilities and consequences of severe nuclear accidents were projected to be low, and health risks to the general public over a long period of time were projected to result primarily from releases of actinide radionuclides during normal operations. No assumptions concerning the environmental transport of transuranic radionuclides or their deposition in human tissues were identified as having sufficient uncertainty to cause more than a factor of two or three changes in the overall projected health effects. The most significant remaining needs are for better information on the physical and chemical nature of released material, environmental weathering of particles, the physical size of particles resuspended from ground surfaces, uptake of radionuclides by plants and the validity of low-dose health-risk extrapolations from high-level biological dose-response studies. In estimating the potential health effects of burning coal, bone tumors and genetic defects were projected to be the major health impact rather than lung tumors, as in the LMFBR simulation. This was the result of the increased importance of the ingestion pathway, although this is probably the most uncertain aspect of the composite model. Still, even the projection for lung tumors was 400 times greater for operation of a coal-fired plant compared to that of an equivalent size LMFBR. This evaluation is limited in scope compared to all the hazards associated with either the total nuclear or coal fuel cycle. It applies the same criteria, degree of conservatism and scientific base of information in making this one health risk comparison.

- 4268 DIFFERENCES IN THE ^{103}Ru AND ^{106}Ru AXIAL DISTRIBUTIONS IN $(\text{U,Pu})\text{O}_2$ FUEL PINS. Phillips, J.R.; Marshall, T.K. (Los Alamos Scientific Lab., NM). Nucl. Technol.; 36: No. 2, 222-228 (Dec 1977).

A precision gamma-scanning system was applied to the nondestructive determination of the isotopic distributions of ^{103}Ru and ^{106}Ru in two failed $(\text{U,Pu})\text{O}_2$ fuel pins irradiated in the Experimental Breeder Reactor II. Significant differences in the two distributions were measured that were related to the half-lives of the two ruthenium isotopes. The two distributions can be used to determine the relative times at which the ruthenium deposits were formed within the fuel material. This information may be used to determine nondestructively the changes within the central void region over a period of several irradiation cycles.

- 4269 INTRUSION OF MOLTEN STEEL INTO CRACKS IN SOLID FUEL IN A TRANSIENT-UNDERCOOLING ACCIDENT IN A LIQUID-METAL FAST BREEDER REACTOR. Ostensen, R.W.; Murphy, W.F.; Wrona, B.J.; Deitrich, L.W.; Florek, J.C. (Argonne National Lab., IL). Nucl. Technol.; 36: No. 2, 200-214 (Dec 1977).

In a transient-undercooling accident in a liquid-metal fast breeder reactor with a small sodium void coefficient, steel cladding melts prior to disruption of the oxide fuel pellets. If molten steel could wet the fuel and penetrate into the interior of the fuel pin through the surface cracks, steel vapor pressure could cause an early breakup and possibly a dispersal of the fuel. The wetting and capillary properties of molten steel in contact with solid oxide fuel have been examined. Based on available indirect in-pile and out-of-pile experimental results, the known surface properties of steel and UO_2 , the

behavior of similar materials, and direct experiments on steel- UO_2 wetting properties, it is concluded that in an accident situation, molten Type 316 stainless steel will not wet oxide fuel at temperatures below the point of disruption of the fuel. Steel intrusions in the fuel will have no significant impact on the accident sequence.

- 4270 DESIGN CONSIDERATIONS FOR LARGE HETEROGENEOUS LIQUID-METAL FAST BREEDER REACTORS. Tzanos, C.P.; Barthold, W.P. (Argonne National Lab., IL). Nucl. Technol.; 36: No. 3, 262-274 (Dec 1977).

A systematic method for designing heterogeneous configurations having a near-zero value of sodium void reactivity is presented. It is based on the following principles: (a) the thickness of the internal blanket zones should be such that the reactivity change resulting from voiding any core zone is practically independent of any further increase in the thickness of these zones, and (b) the sodium void reactivity of each core zone must have a near-zero value. Neutronic coupling among the core zones of heterogeneous configurations decreases as the thickness of the internal blanket zones increases. To quantify coupling, Avery's coupling coefficients are used. Reduced coupling among the core zones of a heterogeneous design, compared to a homogeneous design, results in (a) increased sensitivity of the power distribution to enrichment distribution perturbations, (b) reduced reactivity worth of local perturbations, and (c) higher cladding temperatures during operational transients initiated by local perturbations. Heterogeneous designs compared to equivalent homogeneous designs have (a) lower core Doppler coefficient values, (b) larger fuel compaction reactivities, and (c) higher maximum cladding temperatures.

- 4271 STRESS-FREE SWELLING IN TYPE 304 STAINLESS STEEL AT HIGH FLUENCES. Kenfield, T.A.; Appleby, W.K.; Busboom, H.J. (General Electric Co., Sunnyvale, CA). Nucl. Technol.; 36: No. 3, 347-352 (Dec 1977).

Type 304 stainless steel has been irradiated to a fluence of 1.4×10^{27} n/m², $E > 0.1$ MeV, in the solution-annealed and 10, 20, and 30% cold-worked conditions. Cold working does not impart a continuous reduction in swelling for this alloy. Indeed, in the temperature range from 475 to 550°C (748 to 823 K), the 10 and 20% cold-worked specimens showed more swelling than the alloy in the solution-annealed condition. This behavior appears to be a consequence of differing temperature dependences of swelling. Peak swelling in the cold-worked materials occurs somewhere between 475 and 550°C (748 and 823 K), while the solution-annealed condition peaks at a lower temperature.

- 4272 CREEP-RUPTURE PROPERTIES OF A WELD-OVERLAID TYPE 304 STAINLESS-STEEL FORGING. Klueh, R.L.; Canonico, D.A. (Oak Ridge National Lab., TN). Nucl. Technol.; 36: No. 3, 353-367 (Dec 1977).

The creep-rupture properties of a cylindrical Type 304 stainless-steel forging weld overlaid with Type 308 stainless-steel weld metal were evaluated. Tests were made at 755, 811, and 866 K on tangential and radial (relative to the axis of the forging cylinder) specimens taken from the weld metal, from the forging as forged, and the forging adjacent to the weld metal/forging interface. Two types of axial specimens were also tested: specimens

taken from the unaffected forging and specimens taken across the fusion line. These latter specimens contained both weld metal and forging in the specimen gauge section. At all temperatures, the rupture strength of the forging immediately adjacent to the fusion line was greater than that of the forging specimens taken at a distance from the fusion line. All forging specimens were stronger than the Type 308 stainless steel of the overlay weld metal. The only effect of orientation was found for the overlay weld metal, where the tangential specimens had a slightly greater rupture life than the radial specimens. (The minimum creep rates for the two orientations were similar.) At 811 and 866 K, the ductilities of the overlay weld metal continuously decreased with increased rupture life. A minimum was found for the ductility-rupture life relationship of the Type 304 stainless-steel forging specimens at 866 K, while at 811 K, the ductility continuously decreased with increased rupture life.

- 4273 COMPUTATION STUDIES OF CLADDING RELOCATION DYNAMICS DURING LIQUID-METAL FAST BREEDER REACTOR TRANSIENT UNDERCOOLING ACCIDENTS. Angerer, G. (Kernforschungszentrum, Karlsruhe, Ger.). Nucl. Technol.; 36: No. 3, 305-313 (Dec 1977).

Cladding relocation upon melting has major consequences on the sequence of events in a transient undercooling accident in a liquid-metal fast breeder reactor (LMFBR). The CMOT code developed at the Karlsruhe Nuclear Research Center is used to simulate, by computation, cladding melt-off and blockage formation without and with the possibility of sodium vapor flow diversion. The latter phenomenon is of interest in case of incoherent cladding melt-off within an LMFBR subassembly. It turns out that large waves are generated on the liquid cladding film that quickly slide over a relatively thin slowly moving film. The motion of the waves contributes considerably to the mass transport of cladding film material and to the formation of blockages. The dynamics of these waves is a very important phenomenon of the cladding relocation process. The computed results indicate that cladding blockages in the upper and lower parts of the coolant channel will be established.

- 4274 DYNAMIC INSTABILITIES IN TUBES OF A LARGE CAPACITY, STRAIGHT-TUBE, ONCE-THROUGH SODIUM HEATED STEAM GENERATOR. Unal, H.C.; Gasselt, M.L.G.van (Nijverheidsorganisatie TNO, Apeldoorn-Zuid (Netherlands). Centraal Technisch Instituut TNO); Ludwig, P.W.P.H. (Neratoom N.V., The Hague (Netherlands)). Int. J. Heat Mass Transfer; 20: No. 12, 1389-1399 (Dec 1977).

Density-wave type dynamic instabilities were detected in the tubes of a large-capacity sodium heated steam generator. The peak-to-peak amplitudes of these oscillations varied between 9 to 60 K and the periods between 4 and 7.6s. Experimental data for the periods and the inception conditions of the dynamic instabilities, and the data found in literature obtained for a tube and a rectangular channel across which pressure drop is constant, have been correlated with simple equations for the following range of conditions: pressure: 4.1 to 17 MN/m²; exit quality: 0.27 to 1.59; inlet subcooling: 3 to 111 K; ratio of heated length to hydraulic diameter: 153 to 1921; hydraulic diameter: 0.0045 to 0.0126 m; enthalpy rise in the channel: 582 to 2130 kJ/kg; mass velocity: 353 to 2088 kg/m²s; average heat flux 0.1 to 3.25 MW/m²; period: 0.35 to 7.6s. At the start of dynamic instabilities the steam quality (or

the enthalpy of the steam or steam/water mixture) at the outlet of the boiling channel is constant for a given subcooling, pressure and geometry in the low subcooling range, and for a given pressure and geometry for the range of medium and high subcoolings. The period of the dynamic instabilities is a function of the transit time of a fluid particle in the boiling region of a channel, the Froude number, and the ratio of the length of the boiling region to the total heated length. A hot channel factor has also been established which relates the operating conditions in the tube in which the dynamic instabilities occur first, to the average operating conditions in the steam generator.

- 4275 ARE LMFBR PERMITS UNCONSTITUTIONAL. Wagner, H.; Ziegler, E. Atomwirtsch., Atomtech.; 22: No. 12, 622-627 (Dec 1977). (In German).

The August 18, 1977 decision by the Muenster Higher Administrative Court to have the Federal Constitutional Court investigate the constitutionality of permits granted for fast breeder power plants has aroused much attention, both in the FRG and in other countries. This is the first time that the German Atomic Energy Act is being questioned with respect to the separation of powers between legislative and executive authorities and also with respect to the principle of a constitutional state. As a result of their analysis of the first information available about the court decision the authors have some doubts as to whether the views held by the court about the consequences of the development of fast breeder reactors and of the permit granted for the SNR 300 demonstration nuclear power station are essentially correct. In view of the wording of and the official comments on the incriminated Section 7 of the Atomic Energy Act and the large number of subsequent leading decisions by the Federal Diet about fast breeder reactors also the concern about the constitutionality of that reactor line appears to be unfounded.

- 4276 DISCUSSION OF THE ERDA BREEDER PROGRAM IN THE USA. CRITICAL SURVEY OF THE RELATED PROLIFERATION PROBLEMS. Kessler, G. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Projekt Schneller Bruester). Atomwirtsch., Atomtech.; 22: No. 12, 643-648 (Dec 1977). (In German).

The ERDA breeder program in the United States has been drastically cut back by the Carter Administration. Because of its large reserves of uranium and coal the U.S. feels it is able to postpone considerably the development of commercial fast breeder power plants. This policy is also very much a consequence of the proliferation threat emphasized by the Carter Administration in connection with reprocessing and a plutonium economy. For the same reason, also the other industrialized nations have been requested to rearrange their fast breeder activities accordingly. However, the situation with respect to reserves of primary energy in the German Federal Republic, in the whole of Western Europe and also in Japan compels these countries to make provision for reprocessing spent nuclear fuels, using plutonium as a nuclear fuel and last but not least, the determined development of commercial LMFBR power plants. In addition, an argument used against the new American nuclear policy, both in the United States and in other industrialized countries, is the contention that there is no such thing as an absolutely proliferation-proof fuel cycle and that this

objective could not even be attained by a halt to fast breeder development.

- 4277 SORPTION ON METAL SURFACES OF RADIONUCLIDES FROM SODIUM. Stamm, H.H. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Radiochemie). Atomwirtsch., Atomtech.; 22: No. 12, 639-640 (Dec 1977). (In German).

ALLOYS; BARIUM 140; CHEMISORPTION; CHROMIUM 52; COBALT 60; GETTERING; IMPURITIES; LIQUID METALS; MANGANESE 54; METALS; PRIMARY COOLANT CIRCUITS; PURIFICATION; SODIUM; LMFBTR TYPE REACTORS; COOLANT CLEANUP SYSTEMS; TANTALUM 182; ZINC 65

- 4278 NOTICE OF A PUBLICATION ACCORDING TO ARTICLE 17, PARAGRAPH 1, IN CONNECTION WITH ARTICLE 4, PARAGRAPH 1, OF THE RULES OF PROCEDURE IN NUCLEAR LAW. DATED 15TH DECEMBER 1977. Bundesanzeiger; No. 243, 8 (Dec 1977). (In German).

BUILDINGS; CONSTRUCTION PERMITS; REACTOR LICENSING; SNR-1 REACTOR

- 4279 CONSTITUTIONAL QUESTIONS OF THE DECISION OF THE OVG MÜNSTER CONCERNING THE KALKAR PROCEEDINGS. Backhaus, L. (Münster Univ. (Germany, F.R.). Rechtswissenschaftliches Seminar). Energiewirtsch. Tagesfragen; 27: No. 12, 801-802 (Dec 1977). (In German).

The likely main points of the expected debate are briefly described: 1. The decision relevance within the meaning of art. 100 Basic Law; 2. the substantive-constitutional argumentation of the OVG (Higher Administrative Court), section 7 of the Atomic Energy Act would be a breach of the principle of power sharing (art. 20, section 2, sentence 2 Basic Law).

- 4280 CREEP AND HOT DUCTILITY PROPERTIES OF ALLOY 800 FOR STEAM GENERATOR PIPES OF SURREGENERATORS. Mathern, G.; Seraphin, L.; Tricot, R. (Ugine Aciers, 73 (France)). Rev. Metall. (Paris); 74: No. 12, 717-723 (Dec 1977). (In French).

From 16. Meeting on special steels; Belfort, France (11 - 13 May 1977).

A study was carried out on three castings placed under near-real conditions in the steam generator of the Super Phenix reactor. Measurement of the toughness after holding between 550 and 650°C, creep properties and grain size, and hardness measurement are discussed. The results of electron microscope investigation are given.

- 4281 CAUSTIC STRESS CORROSION OF FE--CR--NI AUSTENITIC ALLOYS. Berge, P.; Donati, J.R.; Prieux, B.; Villard, D. Corrosion; 33: No. 12, 425-435 (Dec 1977).

From 6. international congress on metallic corrosion; Sydney, Australia (Dec 1975).

The susceptibility to stress corrosion cracking (SCC) in caustic solutions of austenitic materials for steam generator tubing, both for liquid metal fast breeders and pressurized water reactors (Type 316, and Alloys 600 and 800) has been compared as a function of: (1) The concentration of the sodium hydroxide solution; (2) the stress level (the actual stress levels for different types of specimens have been carefully evaluated and discussed); and (3) the composition and structure of the metal (in particular, the grain boundary precipitation in high nickel alloys has been characterized by different techniques). It has been shown that the first two factors have a strong influence on the

classification of the alloys, which can be completely reversed when the stress and the caustic concentration change. The possible cracking of Alloy 600 for very low caustic concentrations (4 g/l) suggested the hypothesis that cracking in pure water could be the same phenomenon as caustic cracking. Thermal treatments (such as 16 hours at 700°C) can significantly improve the behavior of Alloy 600 in caustic solutions.

- 4282 STUDY OF MIST FORMATION IN A CONVECTIVE FIELD. EFFECT OF TURBULENT FLUCTUATION ON THE MECHANISM OF MIST FORMATION. Hayashi, Y.; Takimoto, A. (Kanazawa Univ. (Japan). Faculty of Technology); Kanbe, M. Nippon Kikai Gakkai Rombunshu; 43: No. 376, 4598-4607 (Dec 1977). (In Japanese).

Observations were made on the mist formation in natural convective turbulent flow along a vertical plane. The aim of the observations was to clarify the mechanism of mist formation by the critical supersaturation model (CSM) in a turbulent field. Noaphthalene vapor was condensed in the air with sufficient external kernels. Theoretical analysis was also made on the effect of the turbulent fluctuation on the condition of mist formation. The observed experimental results are compared with the results of theoretical analysis in which the fluctuating component is taken into account, and some conclusions are drawn based on the comparison.

- 4283 INELASTIC STRESS ANALYSIS OF STEAM GENERATOR'S TUBE SHEET FOR FAST BREEDER REACTOR. Ishizaki, T.; Maruyama, T. (Babcock Hitachi K.K., Tokyo (Japan)). Karyoku Genshiryoku Hatsuden; 28: No. 12, 1167-1172 (Dec 1977). (In Japanese).

As FBRs and HTGRs are operated in high temperature range where the effect of creep appears conspicuously, the evaluation of ratchet and accumulating inelastic strain taking creep in account is required in the design of equipments, and inelastic stress analysis is required. Babcock Hitachi Ltd. carried out the inelastic stress analysis on the tube plates of steam generators for FBRs. The constructions of a steam generator and the steam exit are shown. For the analysis, the two-dimensional, axis-symmetric, thermo-elasto-plastic creep analysis program, TEPICC-4, was used. This program can make plastic calculation according to tangent rigidity method by the use of four nodes quadrilateral isoparametric elements, the yielding conditions by von Mises, and isotropic hardening law. Besides, the creep calculation is based on initial strain method. The axis-symmetric model of the tube plate part, the division into elements, the conditions of temperature and internal pressure, material constants, and the procedure of calculation are shown. As the result, creep ratchet occurred at the steam side corner of the joint of a tube plate and a shell. At the argon gas side corner, plastic deformation is larger than creep deformation, however, inelastic strain does not come into question because of shakedown. Accumulating inelastic strain satisfied the design standard.

- 4284 SAFETY OF FAST SODIUM-COOLED REACTORS. Bagdasarov, Yu.E.; Kochetkov, L.A.; Matveev, V.I.; Kulakovskij, M.Ya.; Kuznetsov, I.A.; Pomerantsev, G.B.; Krasnoyarov, N.V. At. Energ. (USSR); 43: No. 6, 464-472 (Dec 1977). (In Russian).

On the basis of the analysis of operating experience of fast liquid-metal-cooled reactors considered are the specific features of the

reactor safety and measures of its ensuring, in particular, the application of the emergency cooling systems, designated for significant coolant rates, and the use of specific kinds of electronics for the scram rod control system. The main directions of further developments in engineered safety systems even for the case of unlikely accidents are considered as well. It is shown, that such features of the fast liquid metal-cooled reactors as a great margin before boiling temperature and capability to transfer noninertially the heat from the reactor to other components and heat-exchangers under natural circulation conditions permit to hope, that the risk level for future serial NPPs with fast reactors will not exceed the risk level for the thermal reactor nuclear stations.

- 4285 PRESSURE AND TEMPERATURE STRAINING OF LIQUID SODIUM HEATED SECTIONAL STEAM GENERATORS IN ACCIDENT SITUATIONS. Matel, O. (Vyzkumny Ustav Zavodu Energetického Strojírenství, Brno-Slatina (Czechoslovakia)); Andel, J. (První Brněnská Strojírna (Czechoslovakia)). Jad. Energ.; 23: No. 12, 452-456 (Dec 1977). (In Czech).

The results are presented and discussed of the calculation of the pressure and temperature strain on a sectional steam generator fuelled by liquid sodium and having an output of 30 MW(th). In all sections of the steam generator is the ratio of section length to diameter much greater than 1. In the superheater sections this ratio is 63, in the evaporator and economizer it is 45. The flow rate, pressure, temperature and specific weight of the sodium are taken as constant. For a major leak of water into the sodium in case of a total failure of the heat transfer pipe a typical course is given of the temperature of water/sodium reaction products, the course of wall temperature along the section length filled with reaction products, and the course of the pressure load on the sections and on the steam generator at given parameters (sodium: inlet temperature 445 degC, outlet temperature 292 degC, mean pressure 0.2 MPa; water, steam: inlet temperature 190 degC, outlet temperature 435 degC, mean pressure 8.5 MPa) where the maximum weight flow of water into sodium is $m_{sub(max)} = 3.2 \text{ kg/s}$. An evaluation is made with regard to the project design and the design of the steam generator.

- 4286 REACTOR FUEL CHARGING EQUIPMENT. Wade, E. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). French Patent 2,373,858/A/. 30 Nov 1977. Priority date 7 Dec 1976, United States of America (USA), 17p. (In French).

In many types of reactor fuel charging equipment, tongs or a grab, attached to a trolley, housed in a guide duct, can be used for withdrawing from the core a selected spent fuel assembly or to place a new fuel assembly in the core. In these facilities, the trolley may have wheels that roll on rails in the guide duct. This ensures the correct alignment of the grab, the trolley and fuel assembly when this fuel assembly is being moved. By raising or lowering such a fuel assembly, the trolley can be immersed in the coolant bath of the reactor, whereas at other times it can be at a certain level above the upper surface of the coolant bath. The main object of the invention is to create a fuel handling apparatus for a sodium cooled reactor with bearings lubricated by the sodium coolant and in which the contamination of these bearings is prevented.

- 4287 REACTOR TECHNOLOGY. ATOMIC STRUCTURE.

Erdoes, P. Tech. Rundsch.; 69: No. 48, 9, 11, 13 (29 Nov 1977). (In German).

This is one of a series of articles discussing aspects of nuclear engineering ranging from a survey of various reactor types for static and mobile use to mention of atomic thermo-electric batteries of atomic thermo-electric batteries for cardiac pacemakers. Various statistics are presented on power generation in Europe and U.S.A. and economics are discussed in some detail. Molten salt reactors and research machines are also described.

- 4288 NUCLEAR POWER PLANT WITH IMPROVED ARRANGEMENTS FOR THE REMOVAL OF POST FISSION AND EMERGENCY HEATING. Buescher, E.; Vinzens, K. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,621,256/A/. 24 Nov 1977. 11p. (In German).

This is concerned with additional equipment for emergency heat removal in a sodium cooled reactor, which operates on failure of the post fission heat removal system. The space for pressure relieving spaces and concrete masses as heat sinks within the reactor cell is no longer required. In this nuclear power plant, a heat exchanger chain transmits heat and power: There is a first sodium circuit between pressure vessel and the first heat exchanger, a second one between the first and second heat exchangers, and a third (Steam) circuit with turbine, condenser and return pump. A fourth circuit connects the secondary side of the condenser with a cooling tower. There is a three component heat exchanger in the primary circuit after the first heat exchanger, which is built around the first heat exchanger, and is sealed into an unloading space. This space is situated next to the reactor cell and is above the operating level of the sodium in the pressure vessel. It is connected to the cell by an upper duct, normally closed by a bursting disc, and by a lower duct. In the three component heat exchanger, a liquid lead-bismuth eutectic mixture transmits the heat from sodium pipes to water pipes. In normal operation it is used for steam superheating or feedwater preheating. The three component heat exchanger bridges the first and second heat exchangers as an emergency heat exchanger. If in such a case the post fission heat removal has failed, the sodium evaporating in the pressure vessel flows into the unloading space and condenses on the ribs of the emergency heat exchanger. The post fission heat is fed by the water secondary medium directly into the tertiary circuit. The sodium condensate flows back from the unloading space via the lower duct into the reactor cell and maintains the emergency level there. (RW) 891 RW.

- 4289 INDIRECT HEAT TRANSFER BETWEEN REACTIVE FLUIDS AND DETECTION OF LEAKS INDUCING A MIXTURE BETWEEN THESE FLUIDS. Dickinson, R. (to Babcock and Wilcox Co., New York (USA)). French Patent 2,371,655/A/. 15 Nov 1977. Priority date 17 Nov 1976, United States of America (USA), 10p. (In French).

The invention concerns heat exchangers for indirectly transferring heat between chemically reactive substances or, more generally, mutually incompatible ones (sodium-water exchangers of nuclear reactors). The intermediate capacity formed between the co-axial tubes offers an annular space to a heat conducting substance that is not reactive with the reactive fluids, this intermediate substance serving either as diluent for the fluids arriving in the intermediate capacity

through a leak, or as leak indicator. According to the fluids employed, the presence of a leak can be shown by various physical or chemical occurrences, for instance a change of pressure, temperature, viscosity, colour, odour or electric conductivity.

- 4290 IMPROVEMENTS IN OR RELATING TO SENSORS FOR MONITORING THE ELECTRICAL CONDUCTIVITY OF ELECTRICALLY CONDUCTIVE FLUIDS. McCann, J.D.; Poole, M.J. (to UKAEA). British Patent 1,491,207/A/. 9 Nov 1977. 12p.

A sensor is described that is suitable for monitoring the temperature of the coolant of a liquid metal cooled nuclear reactor or at least variations of the temperature of such a coolant from a predetermined value before it leaves the confines of a fuel sub-assembly and mixes with the outlet streams of other sub-assemblies. The sensor may also be used to detect fluctuations in the flow rate of the coolant. The basis of its action is changes of the electrical conductivity of the coolant with temperature. Constructional details are given. It is pointed out that random mixing of a number of coolant streams at slightly different temperatures produces temperature fluctuations about a mean value and this gives rise to a fluctuating signal sometimes referred to as 'fuel temperature noise', and it is believed that blockage of one coolant stream has a greater relative effect on this fluctuating signal than one on the mean outlet temperature of the coolant stream. Advantages of this type of sensor over conventional thermocouples are stated.

- 4291 NUCLEAR REACTOR FUEL ELEMENT SUB-ASSEMBLIES. Lunt, A.R. (to United Kingdom Atomic Energy Authority). US Patent 4,057,468. 8 Nov 1977. Priority date 3 Mar 1975, United Kingdom of Great Britain and Northern Ireland (UK), 4p.

An elongate fuel element sub-assembly for use in a liquid metal cooled fast breeder nuclear reactor is adapted at a base end region for plugging into core support means and at the upper free end there is inertia damping means to prevent excessive vibration.

- 4292 NUCLEAR REACTOR. Bittermann, D.; Goetzmann, C.; Hassmann, K.; Preuss, H.J.; Rau, P.; Schatz, M. (to Siemens A.G., Berlin (Germany, F.R.)); Siemens A.G., Muenchen (Germany, F.R.)). German(FRG) Patent 2,234,782/B/. 3 Nov 1977. 5p. (In German).

The object of the invention is to ensure a sufficient removal of heat in the case of an accident involving a core melt of a nuclear reactor - in particular a breeder reactor. A cooled tank to catch the melted core is inside the containment under the reactor core. According to the invention, there is a coolant pipeline from this tank to the containment serving as heat exchanger. The cooling of the containment (of metal) used as heat exchanger is further improved upon by a series of measures. 12 subclaims.

- 4293 SECONDARY HEAT TRANSFER CIRCUITS FOR NUCLEAR REACTOR PLANT. Robin, M. (to Commissariat a l'Energie Atomique). US Patent 4,056,439. 1 Nov 1977. Priority date 22 Aug 1975, France, 6p.

A secondary heat transfer circuit for a nuclear reactor comprises a steam generator module, an intermediate heat exchanger, and a pump which circulates a heat transfer liquid, e.g., liquid sodium, between the heat exchanger

and steam generator module. The steam generator module includes a central return duct for cold heat transfer liquid and the pump intake is directly connected to the central duct. In a modification, a plurality of steam generator modules are connected to a single duct.

- 4294 LIQUID SODIUM COOLED FAST REACTOR. Gama, J.M.; Lallement, J.; Pugnet, L. (to Commissariat a l'Energie Atomique). US Patent 4,056,438. 1 Nov 1977. Priority date 18 Oct 1973, France, 6p.

A set of cylindrical shells is interposed between the main vessel wall of an integrated sodium-cooled fast reactor and the primary core containment vessel so as to form a siphon having two downwardly directed branches. One branch opens into the intervessel zone which is separated from the primary vessel by a shell. The other branch which is nearest the main vessel wall is supplied with part of the cold sodium which has been withdrawn from the intervessel zone by means of pumps and reinjected into the bottom of the reactor core.

- 4295 PRODUCTION OF MOLTEN UO_2 POOLS BY INTERNAL HEATING: APPARATUS AND PRELIMINARY EXPERIMENTAL HEAT TRANSFER RESULTS. Chasanov, M.G.; Gunther, W.H.; Baker, L. Jr. (Argonne National Lab., IL). Nucl. Technol.; 36: No. 1, 120-128(Nov 1977).

The capability for removal of heat from a pool of molten fuel under postaccident conditions is an important consideration in liquid-metal fast breeder reactor safety analysis. No experimental data for pool heat transfer from molten UO_2 under conditions simulating internal heat generation by fission product decay have been reported previously in the literature. An apparatus to provide such data was developed and used to investigate heat transfer from pools containing up to 7.5 kg of UO_2 ; the internal heat generation rates and pool depths attained cover most of the ranges of interest for postaccident heat removal analysis. It was also observed in these studies that the presence of simulated fission products corresponding to approximately 150,000 kW-day/kg burnup had no significant effect on the observed heat transfer.

- 4296 SIMULATION OF FAST BREEDER REACTOR PIN BEHAVIOR USING UO_2 PINS WITH DIRECT ELECTRICAL HEATING. Alexas, A. (Kernforschungszentrum, Karlsruhe, Ger.). Nucl. Technol.; 36: No. 1, 129-138(Nov 1977).

In the electrical pin burst facility of the GfK, out-of-pile experiments are performed to simulate the in-pile behavior of fuel pins under SNR-300 accident conditions. Preliminary results of the first experimental series indicate that failure phenomena such as cladding meltdown, fuel melting, and fuel motion, and failure times can be studied by the method of the direct electrical heated UO_2 pin in a satisfactory and reproducible manner. Theory and experimental results will be compared at a later time with the computer program VESUV.

- 4297 CALCULATION OF THE FRAGMENTATION OF UO_2 BY CAPACITOR DISCHARGE: NONEQUILIBRIUM MODEL. Alsmiller, R.G. Jr.; Perez, R.B.; Barish, J. (Oak Ridge National Lab., TN). Nucl. Technol.; 36: No. 1, 139-147(Nov 1977).

A model based on phase-space considerations is developed to describe the fragmentation of UO_2 by capacitor discharge, i.e., to predict such quantities as the amount of gas and liquid produced, the number of liquid fragments, the

number distribution of the molecules in the liquid fragments, the kinetic-energy distribution of the gas and liquid fragments, etc. This model cannot give a unique numerical prediction of all of these quantities based only on the initial-state specification, but it does enable all of these quantities to be expressed in terms of the average internal energy of a gas molecule in the final state, the average binding energy of a UO_2 molecule in a liquid fragment in the final state, and the average number of molecules in a liquid fragment in the final state.

- 4298 SOME ASPECTS OF MODELING POWER PLANT THERMAL HYDRAULIC TRANSIENTS. Bein, M.; Yahalom, R. (General Electric Co., Sunnyvale, Calif. (USA); Fast Breeder Reactor Dept.). Nucl. Eng. Des.; 44: No. 2, 187-192(Nov 1977).

This paper deals with power plant thermal hydraulic modeling, covering the departure from nucleate boiling (DNB), the momentum equation, and the thermal effect of the pipe wall on the fluid heat transport. Two DNB correlations are compared, and the effect of neglecting the change in heat transfer coefficient due to DNB is estimated. The effect of the pipe wall on the thermal dynamics of the flow is analysed.

- 4299 PROBABILISTIC EVALUATION OF CORE-INTERNAL COOLANT FLOW DISTRIBUTION WITH CORRELATION AMONG FUEL SUBASSEMBLIES. Sakai, K.; Hishida, H. (Mitsubishi Atomic Power Industries, Inc., Omiya, Saitama (Japan). Engineering and Development Div.). Nucl. Eng. Des.; 44: No. 2, 235-246(Nov 1977).

The effect of local and global uncertainties in properties and tolerances on the probabilistic variation in coolant flow rate of the hottest channel in a prototype FBR core is discussed in terms of the coolant flow correlation among all the core-internal coolant channels. Sample calculations of the probabilistic deviations in subassembly flow rate for a prototype FBR were about 30% smaller than those conventionally calculated.

- 4300 EXPERIMENTAL STUDY OF WHOLE CORE THERMAL-HYDRAULIC PHENOMENA IN EBR-II. Singer, R.M.; Gillette, J.L. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 44: No. 2, 177-186(Nov 1977).

In the analysis of core-wide temperature distributions in a liquid-metal-cooled fast breeder reactor (LMFBR) under both normal and abnormal operating conditions, it is commonly assumed that there is little, if any, thermal interaction between adjacent subassemblies. Since intersubassembly heat transfer tends to reduce the transverse temperature gradients in a reactor, thereby ameliorating the effects of local overheating, this assumption is conservative. In order to assess the importance of this effect as well as of flow redistribution in a reactor core, an experimental study was conducted in EBR-II covering a wide range of operating power and flow conditions, including both forced and natural convection. The results of this study indicate that radial heat transfer and flow redistribution are important mechanisms in the thermal-hydraulics of LMFBR cores, especially at low flow rates.

- 4301 INTERNAL PRESSURIZATION IN SOLID MIXED-OXIDE FUEL DUE TO TRANSIENT FISSION GAS RELEASE. Hofmann, J.R.; Meek, C.C. (Argonne National Lab., Ill.). Nucl. Sci. Eng.; 64: No. 3, 713-723(Nov 1977).

A model employing Darcy's law has been

developed to describe the transient pressure field within interconnected porosity of mixed-oxide liquid-metal fast breeder reactor fuel during hypothetical reactor accidents. Pressure increases are due both to fission gas released from fuel grains and fill gas originally present within fuel pores. Calculations utilizing the model have been performed for an out-of-pile test prior to fuel melting with both clad and unclad conditions being treated. Redistribution of gas from the source region in the relatively high-porosity unstructured fuel to a low-porosity restructured fuel was shown to exist in all cases considered. Even for the unclad case, significant internal pressurization was predicted by the model, which could prove important in subsequent fuel breakup and motion.

- 4302 SCALING UP THE FAST BREEDER. Nucl. Eng. Int.; 22: No. 264, 68-69(Nov 1977).

A brief description is given of the British experience in operating the PFR at Dounreay and plans for scaling up the prototype to a commercial demonstration fast reactor (CDFR). Major components performed extremely well in Na. Experience with leaks in steam generators and superheaters has results in some design changes for CDFR. Comparison of main parameters shows that the larger output of CDFR is provided by a greater number of components so that extrapolation in component size is generally kept small. Lower temperature will reduce thermal stress problems, and will also allow the use of 9% Cr steel superheaters. Primary containment will be prestressed concrete. The problem of security in relation to transportation of Pu is discussed.

- 4303 MAINTAINING AN EFFECTIVE BREEDER REACTOR PROGRAMME. Beckford, E.S. (Dept. Energy, (US) Division of Reactor Development). Nucl. Eng. Int.; 22: No. 264, 55-57(Nov 1977).

The near-term and long-term objectives of the US advanced nuclear programme are summarised and the Non-Proliferation Alternative System Assessment Program (NASAP) is discussed. US work on the sodium breeder is being restructured to examine alternate fuel cycles which minimise or eliminate the use of Pu. The test facilities and areas of technology involved are considered.

- 4304 SEISMIC DESIGN STUDY ON CANTILEVER CYLINDRICAL SHELLS. Wu, J.N.C.; Cory, J.E. Jr. (Babcock P Wilcox Co, Akron, Ohio). J. Pressure Vessel Technol.; 99: No. 4, 600-604(Nov 1977).

Paper No. 77-PVP-14.

- 4305 IN-REACTOR STRESS RELAXATION OF INCONEL X750 SPRINGS. Walters, L.C.; Ruther, W.E. (Argonne National Lab., Idaho Falls, Idaho (USA)). J. Nucl. Mater.; 68: No. 3, 324-333(Nov 1977).

In 1965 eight surveillance subassemblies were placed in row 12 of the EBR-II sodium-cooled fast breeder reactor with an irradiation temperature near the sodium-inlet temperature of 371°C. At the same time, two other surveillance subassemblies were placed in the primary storage basket, which receives minimal neutron exposure but is immersed in primary sodium and experiences a temperature of 371°C. Each of the subassemblies contained 18 preloaded springs made of Inconel X750. Springs from four of the in-core subassemblies and one subassembly from the storage basket have been evaluated to determine irradiation-enhanced deformation rates to neutron exposures of 4.2

dpa. It was found that the creep coefficient derived from the stress relaxation measurements on Inconel X750 springs was 1.0×10^{-12} (Pa-dpa)⁻¹ for springs irradiated up to 4.2 dpa (3751 d) at an in-reactor temperature of 371°C. The relaxation behavior was adequately described by a creep law that was linear in neutron fluence and applied stress. Springs encapsulated in helium showed identical in-reactor relaxation rates to springs exposed to the flowing primary sodium. The creep coefficient derived from the present work on Inconel X750 springs was shown to be the same as the creep coefficients determined from various austenitic steel alloys.

4306 BEHAVIOR OF ALLOY 800 IN SODIUM

CONTAMINATED BY SODA. Cappelaere, M.; Dixmier, J.; Sannier, J.; Coriou, H. (CEA, Cent d'Etud Nucl de Fontenay-aux-Roses, Fr). Mem. Sci. Rev. Metall.; 74: No. 11, 719-730 (Nov 1977). (In French).

A water leak in one of the steam generators in a nuclear plant cooled with sodium may lead to formation of soda (in addition to other compounds). The harmful effect of soda on the austenitic Fe-32Ni-21Cr alloy, called alloy 800, used in the steam generators of the Superphoenix reactor was therefore investigated. Stress corrosion tests showed that with high soda contents of 8% and more, the risk of stress corrosion cracking is real. With 16% soda under a stress of 130 N/mm² sup 2s the specimen fractured in 20 hr. With small amounts of soda (of about 1%) under the same stress, no fracture was observed after 100 hr.

4307 NUCLEAR POWER PLANTS OF THE SOVIET

UNION. Ermakov, G.V. (State Comm for Sci and Tech, USSR). Teploenergetika (Moscow); No. 11, 47-53 (Nov 1977). (In Russian).

Detailed description and design diagrams of several types of Soviet reactors are given, e.g., the uranium-graphite channel reactors with aqueous heat carrier installed at the Beloyarskaya nuclear power plant; its successor, the water-graphite boiling reactors of the channel type (RBMK-1000), installed at the Leningrad nuclear power plant; its further development, the RBMK-1500 presently being installed at the Ignalinskaya nuclear power plant; the uranium-graphite reactors with boiling water heat removal of the EGP-6 type installed at the Bilibinskaya nuclear heat-and-power plant (Magadan region); pressurized light water reactors of the VVER type, installed at the Novovoronezhskaya nuclear power plant, etc. Steam generators for power plants with VVER-440 and VVER-1000 reactors are described. The fast neutron breeder reactor, BN-350, installed at the city of Shevchenko in 1973 is briefly described and its design diagrams are given.

4308 DEVELOPMENT OF NUCLEAR POWER ENGINEERING

IN THE USSR. Karelin, E.P.; Sukhov, A.B. Teploenergetika (Moscow); No. 11, 54-59 (Nov 1977). (In Russian).

Brief characteristics of the first commercial Soviet nuclear power plants are presented. In particular, descriptions of the power units at the Novovoronezhskaya, Beloyarskaya, Armenian, and Shevchenko nuclear power plants are given. It is stated that, in the long run, nuclear power plants will mainly be built in the unified power systems of the North East, Center, South, and Middle Volga of the European part of the USSR. Specific costs of nuclear power plants have been reduced to 200-210 roubles/kW, which is by about 45% higher than the costs of ordinary steam power plants of the same capacity. Prime costs of

electric power produced by them are lower, however. A table of the existing nuclear power plants in the USSR is given.

4309 LOCAL SODIUM BOILING BEHIND LOCAL FLOW BLOCKAGE IN SIMULATED LMFBR FUEL SUBASSEMBLY.

Kikuchi, Y.; Daigo, Y.; Ohtsubo, A. (Power React and Nucl Fuel Dev Corp, Ibaraki-ken, Jpn). J. Nucl. Sci. Technol. (Tokyo); 14: No. 11, 774-790 (Nov 1977).
The paper deals with local sodium boiling in the downstream of a six-subchannel blockage in an electrically heated LMFBR fuel subassembly mock-up. The first series of experiments were conducted to measure temperature distributions in the downstream of the blockage under nonboiling conditions. The measured temperature rise due to the blockage agreed fairly well with the calculation by the LOCK code. The second series of experiments were performed to investigate local boiling phenomena. In the local boiling region, no flow instability was observed since the subchannels near the wrapper wall were still filled with subcooled liquids. The boiling caused a considerable increase in acoustic noise intensity. The frequency (2.9 and 20.2 sec/sup -1s) of bubble formation decreased with the increase of the bubble size at its point of maximum development. The product of the bubble frequency and the equivalent diameter was found to be constant. 7 refs.

4310 (JAPFNR--296) TEST ON "MONJU" FUEL SUBASSEMBLIES (THE FIRST AND THE SECOND TEST FABRICATIONS).

Kamimura, K.; Uematsu, K.; Sato, K. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Nov 1977. Translation of N--241-77-32, December 1976 137p. Dep. NTIS (US Sales Only), PC A07/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.
This report covers the details of "the first and the second test fabrications of MONJU fuel subassemblies," "water flow test," "sodium flow test," and "post sodium flow examination."

4311 THERMO-HYDRODYNAMIC INSTABILITY TEST FOR 0.3 MWT SODIUM-HEATED STEAM GENERATOR.

Harada, T.; Tomita, Y.; Yano, T. (Ishikawajima-Harima Heavy Industries Co. Ltd., Tokyo (Japan)). Ishikawajima-Harima Giho; 17: No. 6, 558-565 (Nov 1977). (In Japanese).
In order to obtain information respecting the water-side instability in sodium-heated steam generators, the conditions to induce the instability were investigated on a 0.3 Mwt test model. Firstly, the model was confirmed to be entirely stable for the range of normal operating conditions. Secondly, it was judged from the oscillation characteristics of the flow instability that the instability was due to density wave oscillation. Thirdly, through a series of experiments where the flow instability was forcibly induced by throwing open the inlet orificing valve widely, the data for the steady-state static heat transfer just before the onset of instability and the data for the stability/instability threshold were obtained. Further, a computer program to predict the static performances was developed, and the heat input data thereunder computed were fed to the DYNAM Code to analyze the dynamic stability. As a result of the comparison with the experimental data for the stable operation, the DYNAM Code proved to have a fair amount of safety margin.

- 4312 LOCAL SODIUM BOILING BEHIND LOCAL FLOW BLOCKAGE IN SIMULATED LMFBF FUEL SUBASSEMBLY. Kikuchi, Y.; Daigo, Y.; Ohtsubo, A. (Power Reactor and Nuclear Fuel Development Corp., Darai, Ibaraki (Japan)). Darai Engineering Center). J. Nucl. Sci. Technol. (Tokyo); 14: No. 11, 774-790 (Nov 1977).

This paper deals with local sodium boiling in the downstream of a six-subchannel blockage in an electrically heated LMFBF fuel subassembly mock-up. The first series of experiments were conducted to measure temperature distributions in the downstream of the blockage under non-boiling conditions. The measured temperature rise due to the blockage agreed fairly well with the calculation by the LOCK code. The second series of experiments were performed to investigate local boiling phenomena. In the local boiling region, no flow instability was observed since the subchannels near the wrapper wall were still filled with subcooled liquid. In the nearly bulk boiling region, however, considerable upstream voiding occurred and then the inlet flow decreased, leading to final dryout. The boiling caused a considerable increase in acoustic noise intensity. The root-mean-square (RMS) noise level of approximately 20 mbar obtained in the present local boiling experiments with sodium was much higher than that (approximately 0.5 mbar) in the ordinary nucleate boiling experiments with water. The peak observed in the hertz ranges was due to the repetition of bubble formation and collapse. In the kilohertz ranges, however, resonance peaks were superposed on a smooth curve with a broad peak at approximately 7 kHz. The frequency (2.9 and 20.2 sec⁻¹) of bubble formation decreased with the increase of the bubble size at its point of maximum development. The product of the bubble frequency and the equivalent diameter was found to be constant.

- 4313 STATUS OF THE RESEARCH AND DEVELOPMENT PROGRAM ON FAST REACTORS IN THE USSR. Kazachkovskii, O.D.; Meshkov, A.G.; Mitenkov, F.M. . At. Energ. (USSR); 43: No. 5, 343-351 (Nov 1977). (In Russian).

The development of LMFBF type reactors in the USSR is shortly reviewed beginning with the BR-10 reactor. Operating experience of the BR-10 MW(th), BOR-60 MW(th), and BN-350 MW(e) reactors is summed up. Basic technical characteristics of the reactors as well as of the BN-600 MW(e) reactor (under construction) and the BN-1600 MW(e) reactor (planned) are compared. The designs of the BN-1600 reactor with a 'vertical' and 'horizontal' scheme integrating the core and the primary cooling system equipment within a single tank are discussed at greater length. The performance of circulating sodium pumps and sodium-water heat exchanges in the operating reactors is given. Oxide fuels are treated as the basic variant to pursue, but carbide, nitride and metallic fuels are also considered as likely candidates. Shortening of the fuel recycle duration is considered as one of the vital problems to tackle. It is pointed out that apart LMFBF type reactors other fast breeder reactors with He and dissociating gaseous N₂O₄ coolants are under investigation.

- 4314 ELECTROMAGNETIC PUMP FOR FAST BREEDER REACTOR. Nanba, M.; Mukaida, H.; Miyagi, T. (Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Ryutai Kogaku; 13: No. 11, 666-673 (Nov 1977). (In Japanese).
As for the circulating pumps for fast breeder reactors, mechanical pumps are employed for large flow rate, and electromagnetic pumps for small flow rate. The electromagnetic pumps

are inferior in pumping efficiency, but superior in safety and reliability to the mechanical pumps. The electromagnetic pumps are classified into two types: the conduction type to electrify the fluid directly from the exterior and the induction type to induce current in the fluid with a moving magnetic field. The induction type is more reliable as compared with the conduction type because of its simple duct construction. The simple theory for the induction type pumps is described, and three models with different constructions and features are explained, namely flat linear induction pump, annular linear induction pump, and helical induction pump. For putting fast breeder reactors to practical use, the electromagnetic pumps are required to realize the improvement of safety and reliability, high performance, compactness and light weight, the improvement of operation control, and easier installation and maintenance. The electromagnetic pumps may be adopted as the main circulating pumps for fast breeder reactors if the higher efficiency including thermal input is realized. For this purpose, the annular linear induction pumps are promising.

- 4315 ECONOMIC ASPECTS OF THE DEVELOPMENT OF NUCLEAR POWER AND FUEL CYCLE INDUSTRY IN THE USSR. Dergachev, N.P.; Kruglov, A.K.; Sedov, V.M.; Shuklin, S.V. . At. Energ. (USSR); 43: No. 5, 365-369 (Nov 1977). (In Russian).

The likely growth rate and technological structure of nuclear power generating capacities in the USSR have been investigated as well as the distribution of capital investments into NPP's and nuclear fuel cycle industries. The added NPP capacities are assumed to consist entirely of NPP's with the WWR -1000 MW(e), 1000 MW(e) LWGR type and BN-1600 MW(e) reactors. Basic formulas are given to calculate the projected growth rate of total nuclear capacity and NPP's with breeder reactors. The plutonium supply in the near 20-25 years is concluded to hardly restrain the development of NPP's with breeder reactors which would reduce the natural uranium demand by about 25%. The major flow of capital is required to construct NPP's and only about 20% of the total is invested into the fuel cycle industry. Hence, reduction of NPP capital costs is the most desired and effective undertaking. Uranium ore mining and milling account for as much as half of fuel cycle costs. Therefore, introduction of uranium and plutonium recycle and reduction of tails assay at enrichment plants are concluded to be the most efficient means to cut fuel costs.

- 4316 EFFECT OF FAST NEUTRON IRRADIATION ON THE STRUCTURE OF BORON CARBIDE. Hollenberg, G.W.; Cummings, W.V. (Westinghouse-Hanford Co., Richland, WA). J. Am. Ceram. Soc.; 60: No. 11-12, 520-525 (Nov 1977).

The leading LMFBF control rod material, B₄C, was examined by SEM, TEM, and X-ray diffraction after irradiation up to 80 x 10²⁰ captures/cm³ between 500° and 920°C. Bulk swelling was found to be the result of elastic and plastic strains, rather than dilation of the unit cell. Microcracking was observed and explained in terms of anisotropic swelling.

- 4317 ANALYTICAL METHOD OF FUEL PIN DEFORMATIONS IN LMFBF ASSEMBLY. Miki, K. (Hitachi Ltd., Kawasaki, Kanagawa (Japan)). Atomic Energy Research Lab.). J. Nucl. Sci. Technol. (Tokyo); 14: No. 11, 791-804 (Nov 1977).

A new analytical method is presented for

analyzing, in three dimensions, the mechanical response of fuel pins with wire spacers, to their thermal and neutronic environment in a fuel assembly of a LMFBR. It analyzes the mechanical interactions between fuel pins in the assembly in each of three directions, which form an angle of $\pi/3$ radians with one another, based on the mathematical relationship between the displacements at the contact points and the associated contact forces with respect to all fuel pins forming a line in one of the three directions. Based on this method, a new computational code, the Subchannel Deformation Analysis Code for Wire-Wrap Assemblies (SHADOW) has been developed, and is applied to a fuel assembly of a prototype fast breeder reactor in order to analyze the deformation of 169 fuel pins due to thermal bowing. Conclusions drawn from the study confirm that the SHADOW code can be an effective tool for analyzing or evaluating thermal and structural designs of a LMFBR fuel assembly.

- 4318 ELEVATED-TEMPERATURE LOW-CYCLE FATIGUE BEHAVIOR OF DIFFERENT HEATS OF TYPE 304 STAINLESS STEEL. Maiya, P.S.; Majumdar, S. (Argonne National Lab., IL). Metall. Trans., A; 8: 1651-1660 (Nov 1977).

The low-cycle fatigue results of three heats of Type 304 stainless steel have been obtained at 593°C under selected cyclic-loading conditions. The results are compared with those generated for a reference heat of steel for which extensive low-cycle fatigue data are available. Observation of the microstructures of specimens in the pretest condition after a given heat treatment and examination of fatigue fracture surfaces were conducted by means of optical and scanning electron microscopy and X-ray analysis. The three heats of stainless steel, which exhibit different microstructural features, show approximately the same continuous-cycling low-cycle fatigue behavior as that of the reference heat. However, the three materials show improved fatigue strength during tensile hold-time conditions where significant creep occurs. The fatigue properties determined in the present study for the different heats of steel are consistent with the observed microstructural features. Finally, the creep-fatigue properties of the heats as well as the microstructural observations are discussed in terms of a damage-rate approach recently developed by the authors.

- 4319 (DOE-tr--69) TESTING OF A GRAPHITE-BASED EXTINGUISHING POWDER FOR USE ON LIQUID METAL FIRES. Menzenauer, P.; Ochs, G.; Peppler, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Nov 1977. Translation of KFK--2525 20p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States--W. Germany Fast Reactor Exchange Program.

A graphite-based extinguishing powder, newly developed by a French firm (CECA S.A.--Le Carbone Lorraine) for use on liquid metal fires has been tested on a sodium fire in a pan with a surface area of 0.96 m² using 60 kg of sodium. The behavior and extinguishing power are reported and compared with those of other materials tested in earlier experiments. The new powder has excellent efficiency in dealing with sodium pool fires. Application is simple. The amount required is more than one order of magnitude less than that of currently available alternatives. The powder is not corrosive.

- PLANT. Doryokuro Giho; No. 24, 72-98 (Nov 1977). (In Japanese).

The design of the prototype fast breeder reactor "Monju" has been in progress since 1968. "Monju" is a 300 MWe, loop type plant with fuel of mixed oxides of plutonium and uranium. The core is divided into two zones with different plutonium enrichment, and composed of hexagon-shaped fuel subassemblies. The coolant temperatures at reactor inlet and outlet are 397 deg. C and 529 deg. C, respectively. The expected average fuel burn-up for some early stage of operational period is 55,000 MWd/t. Heat is removed from the reactor core and blanket by the primary sodium coolant, transferred to the secondary sodium coolant in three parallel intermediate heat exchangers, and finally transferred to water and steam in three once-through type steam generators composed of helical coil tubes, and generates superheated steam of 487 deg. C at 132 kg/cm² g which is used for a turbine. Decay heat removal is normally accomplished by means of three parallel auxiliary core cooling systems. The plant uses a top-supported reactor vessel of 7.1 meter in diameter and 17.8 meter in height, which is closed at the top with a shield plug. The reactor is equipped with 19 control rods, and B₄C is used as the neutron absorber. The fuel handling machine is a fixed arm type. The reactor containment consists of the primary and secondary boundaries to isolate the radioactivity released in a hypothetical accident. For the reactor building and the auxiliary service building, high reliability and integrity are required from seismic standpoint. The reactor building is located at the center of the single solid foundation which also supports the auxiliary service building surrounding the reactor building.

- 4321 BEHAVIOR OF ALLOY 800 IN SODIUM CONTAMINATED BY SODIUM HYDROXIDE. Cappelaere, M.; Dixmier, J.; Sannier, J.; Coriou, H. (CEA Centre d'Etudes Nucleaires de Fontenay-aux-Roses, 92 (France). Div. de Chimie). Mem. Sci. Rev. Metall.; 74: No. 11, 719-730 (Nov 1977). (In French).

From 16. Meeting on special steels; Belfort, France (11 - 13 May 1977).

The behavior of various metallic materials in NaOH-contaminated sodium was studied in the temperature range 475-550°C. It was observed that the presence of NaOH in sodium induced a generalized corrosion of ferritic steels while austenitic material were subjected to intergranular penetrations even without stress. In a second part, alloy 800 was specially investigated, since it was chosen as a standard material for Superphenix steam generators. Parametric tests were carried out under tensile stress to study susceptibility to cracking of this alloy. A device for introducing NaOH into sodium at the experiment temperature was specially designed. The results show that, for high concentration of NaOH (8 weight % and more), there is a real risk of stress cracking. So ruptures are observed in samples which show or not welded areas after 20 hours with 16% of NaOH under 130 N.mm⁻² stress. On the other hand, for low initial concentrations (about 1%) and with the same stress conditions, no cracking occurs after 100 hours. Finally surface condition of material has an effect on the formation of the oxidized surface layer but this parameter has no significant influence on the sample life-time.

- 4322 DEVICE FOR THERMAL INSULATION OF A VESSEL WALL. Lemerrier, G. (to Commissariat a l'Energie Atomique). US Patent 4,055,465. 25 Oct 1977. Priority

date 24 Jul 1975, France, 6p.

A thermal insulation structure mounted within the annular space between the primary vessel and the inner vessel of a liquid metal cooled fast reactor is constituted by a plurality of heat-insulating elements extending circumferentially around the common axis of the two vessels and consisting of half-casings each formed by two interengaged L-section members packed with flexible and compressible heat-insulating material.

- 4323 LIQUID METAL COOLED REACTOR AND PROCESS FOR PROTECTING THE CHANNELS AGAINST A BLOCKING. (to UKAEA London Office). French Patent 2,368,119/A/. 17 Oct 1977. Priority date 18 Oct 1976, United Kingdom of Great Britain and Northern Ireland (UK), 12p. (In French).

This invention concerns the nuclear fuel for liquid metal cooled nuclear reactors comprising a bundle of parallel nuclear fuel pins, very close together encased in an annular cladding. Under the invention, a liquid cooled nuclear reactor includes a core with a channel in which the coolant flows along subsidiary channels fitted between parallel fuel pins to extract the heat from them and a transducer sensitive to a medium temperature at the end of the channel outlet to monitor the correct cooling of the fluid. A certain number of heat transmission pipes open at the ends are distributed along the flow section of the channel to extract the heat from any localised area of the coolant flowing in the subsidiary channels, the resulting temperature rise of which is not communicated in any other manner to the transducer by the coolant flowing in the subsidiary channels in contact with the fuel pins. The above mentioned localised areas are those in which the coolant is locally detached from the main current flowing in the channel. The invention also concerns a process for monitoring a channel reserved for a liquid coolant in the above mentioned manner, as well as a nuclear, cooled by a liquid metal, fitted with the above mentioned device making it possible to detect the temperature prevailing in a coolant channel.

- 4324 LIQUID METAL COOLED NUCLEAR REACTORS. Barnes, S.; Fothergill, J.R. (to UKAEA). British Patent 1,488,478/A/. 12 Oct 1977. 6p.

Reference is made to liquid metal cooled nuclear reactors of the 'pool' type. In such reactors the core, the heat exchangers, and the coolant circulating pumps are submerged in a pool of liquid metal. In operation of the reactor it is necessary to be able to locate and identify components submerged in the pool, and before moving rotating shields in the roof of the pool-containing vault it is necessary to ensure that all the normally suspended absorber rods have been inserted in the core and released from their suspensions. Television cameras are unsuitable for use in the opaque liquid metal but ultrasound in the megahertz range has been used to give a television screen kind of display. There is some difficulty, however, in transmitting ultrasound signals from a transducer into the pool of coolant because the transducer must be protected from the high temperature environment of the coolant. This difficulty has been partially overcome, however, by transmitting the signals by way of a wave guide extending from the transducer into the coolant pool. Such a wave guide may comprise a column of liquid metal within a dip tube. The column of liquid coolant is uninterrupted by a supporting diaphragm. Such a system is here described.

- 4325 (DOE-tr--70) FR2-LOOP EXPERIMENTAL SERIES 5, IRRADIATION, POST-IRRADIATION EXAMINATION, AND ANALYSIS OF THE FCMI. Mueller-Lyda, I. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). 10 Oct 1977. Translation of KFK--2523 54p. Dep. NTIS (US Sales Only), PC A04/MF A01.

Work performed under United States--W. Germany Fast Reactor Exchange Program.

In the FR2-Loop experimental series 5, the plastic cladding tube diametral strain of UD_2 -- PuO_2 test rods was examined after alternating load operation. The pin design was different from the SNR-300 specification with respect to a strong FCMI (Fuel-Cladding Mechanical Interaction). The irradiation was carried out in a high number of single cycles (300 to 500 W/cm linear power, 1 to 600 h duration) with a subsequent diameter measurement. It turned out that within the attained burnup range (< 2%) no considerable diameter increases occurred, even with very rough cycling, except one case with a very special power and temperature history.

- 4326 METHOD AND APPARATUS FOR LOCATING A DEFECTIVE TUBE OF A LIQUID METAL-TO-WATER TUBE TYPE HEAT EXCHANGER. Schrock, S.L.; Mangus, J.D. (to Westinghouse Electric Corp.). US Patent 7963,003. 4 Oct 1977. Filed date 12 Apr 1976. 4p.

Disclosed is a method and apparatus for detecting a specific leaking tube or tubes and for detecting the position of a defect within a tube in a liquid metal-to-water tube type heat exchanger, generally used in a sodium cooled nuclear reactor. Subsequent to draining of the heat exchanger a solid reaction product of the liquid metal and water forms at a leak location. Introducing an inert gas to the exterior of the tubes, creating a pressure differential across the tube wherein the interior pressure is less than the exterior pressure, and heating the tube, provide dissociation and isolation of the reaction product, which, when analyzed for chemical content, indicates which tube or tubes is defective. A heat probe apparatus which traverses the interior of the tubes may be used in the heating step, which, as a result of the speed of the dissociation process at high temperatures provides a means for detecting the specific leak location within the tube. A heat apparatus which traverses the exterior of bayonet type tubes may be used in another embodiment.

- 4327 GENERALIZED PARAMETRIC MODEL FOR TRANSIENT GAS RELEASE AND SWELLING IN OXIDE FUELS. Gruber, E.E. (Argonne National Lab., IL). Nucl. Technol.; 35: No. 3, 617-634 (Oct 1977).

The need for a capability to predict transient fission gas behavior arises because of the complexity of fission gas response to transient conditions, the importance of fission gas to fuel mechanical response, and the prevalent limitations on experimental information relevant to the problem. A detailed mechanistic analysis of intragranular swelling and release of gas from grains to grain boundaries, both as they result from transient heating, has been developed and incorporated in a fission gas release and swelling code (FRAS). A generalized parametric model to approximate the results that would be obtained from the more detailed calculations has also been developed. The need for this model arises from the necessity to consider the fission gas effects in more general multinode accident-analysis and pin-mechanics codes. For such

calculations, the FRAS code is prohibitive in its demands on computer storage and execution time, while the parametric FRAS (PFRAS) code reduces these demands by an order of magnitude. Transient calculations have been carried out with both codes, both to illustrate the sensitivity of the results to the parameters and to indicate the level of confidence that can reasonably be ascribed to PFRAS results. The parameters considered include initial gas concentration, grain size, heating rate, thermal gradient, and pressure. The PFRAS model gives a satisfactory approximation to FRAS results for the broad range of parameters surveyed.

been introduced to provide greater flexibility. The details of a direct-coupled-ray (DCR) + P_0 conversion technique have been developed but not yet implemented. The DCR method, as implemented in the TRANS3 code, has been used in a number of liquid-metal fast breeder reactor shielding applications. These included a one-dimensional deep penetration configuration and one-, two-, and three dimensional representations of the lower axial shield of the Clinch River Breeder Reactor. Comparisons with ANISN and DOT-III solutions indicated good to excellent agreement in most situations.

- 4328 INTERACTIONS OF REFRACTORIES AND REACTOR MATERIALS WITH SODIUM. Fink, J.K.; Heiberger, J.J.; Kumar, R.; Blomquist, R.A. (Argonne National Lab., IL). Nucl. Technol.; 35: No. 3, 656-662(Oct 1977).

As part of a program at Argonne National Laboratory (ANL) to investigate the compatibility of high-temperature sodium with materials being considered for core retention systems in liquid-metal fast breeder reactors, various commercial refractories and samples of reactor control materials were exposed to static sodium at 850°C for 5 h. The refractories tested were samples of magnesia, alumina, zirconia, mixed ceramic oxides, and graphite; the reactor control materials were boron carbide and tantalum. Samples of graphite, zirconia, and the refractories with high alumina or magnesia contents, but with low silica and chromic oxide contents, were found to be compatible with high-temperature sodium. Sample compatibility with sodium decreased with an increase in the silica content of the sample. Samples with large silica content failed completely. These results are in good agreement with results of other experiments, performed at ANL and at the Westinghouse Advanced Reactors Division, in which these materials were exposed to boiling sodium.

- 4329 ELASTIC--PLASTIC ANALYSIS OF PRESSURIZED DUCTS WITH ROUNDED CORNERS. Petroski, H.J. (Argonne National Lab., IL). Nucl. Technol.; 35: No. 3, 671-676(Oct 1977).

The strength of a reactor subassembly duct depends on its cross section and its material properties. Straight- and curved-beam theories have been used to analyze ducts with elastic-perfectly-plastic material behavior, and this analysis has been found to be suitable and convenient for parametric studies. The strengths of hexagons with different corner radii have been compared, and the stiffening effect of increasing radius has been quantified. The relative strengths of different duct cross sections have also been determined.

- 4330 DIRECT-COUPLED-RAY METHOD FOR DESIGN-ORIENTED THREE-DIMENSIONAL TRANSPORT ANALYSIS. Bucholz, J.A. (Oak Ridge National Lab., TN); Poncellet, C.G. Nucl. Sci. Eng.; 64: No. 2, 356-372(Oct 1977).

A fast three-dimensional design-oriented transport method has been developed for the solution of both neutron and gamma transport problems. It combines a nodal approach with analytic integral transport to achieve relative speed and accuracy. An analytic solution is obtained for the angular flux in each of the 14 directions defined by the six faces and eight corners of a cubic mesh block. The scheme used to accommodate high-order anisotropic scattering is based on the formulation of ray-to-ray scattering probabilities in an integral sense. A variable mesh approximation has also

- 4331 OPTIMIZED ITERATION STRATEGIES AND DATA MANAGEMENT CONSIDERATIONS FOR FAST REACTOR FINITE DIFFERENCE DIFFUSION THEORY CODES. Ferguson, D.R.; Derstine, K.L. (Argonne National Lab., IL). Nucl. Sci. Eng.; 64: No. 2, 593-604(Oct 1977).

Optimized iteration methods for the solution of large-scale fast reactor finite difference diffusion theory calculations are presented, along with their theoretical basis. The computational and data management considerations that went into their formulation are discussed. The methods utilized include a variant of the Chebyshev acceleration technique applied to the outer fission source iterations and an optimized block successive overrelaxation method for the within-group iterations. The performance of a computer code employing these methods is compared with that of several existing production diffusion theory codes for a range of typical problems.

- 4332 (ANL-Trans--1106) AUTONOMOUS DEVICE FOR AFTER-HEAT REMOVAL FROM A SODIUM COOLED FAST NEUTRON REACTOR. Oct 1977. Translation of French Patent Application No. 7,335,223 11p. Dep. NTIS, PC A02/MF A01.

An autonomous device is described for removal of the after-heat in a sodium-cooled fast reactor of the type in which each cooling loop comprises the primary sodium circuit exchanging its heat with the sodium of a secondary circuit through an intermediate exchanger, characterized by the fact that each cooling loop in parallel with the secondary sodium circuit contains an auxiliary secondary circuit in which a chemically inert gas draws heat from the primary circuit, said auxiliary secondary circuit comprising at least one gas turbine fed by said inert gas and operating in an energy-generating closed cycle.

- 4333 PHYSICS OF LIQUID METAL FAST BREEDER REACTOR SAFETY. Wilson, R. (Department of Physics, Harvard University, Cambridge, Massachusetts 02138). Rev. Mod. Phys.; 49: No. 4, 893-924(Oct 1977).

The main physical features of LMFBR safety are summarized with particular reference to the maximum possible mechanical energy release in a hypothetical core disruption accident. It is likely that such a postulated fuel melting accident would lead to a nonenergetic disassembly of the core. With pessimistic estimates of reactivity increases, but not derived from any mechanistic point of view, mechanical energy impacts on the reactor vessel of 100 MJ have been calculated for the Clinch River Breeder Reactor. These are containable with the present design, and the increase should vary only linearly with increased power for future designs of similar concept. Various possible steps are discussed to increase the certainty of these conclusions.

- 4334 MATHEMATICAL AND COMPUTATIONAL TECHNIQUES EMPLOYED IN THE DETERMINISTIC APPROACH TO LIQUID-METAL FAST BREEDER REACTOR SAFETY. Walter, A.E. (Univ. of Virginia, Charlottesville); Padilla, A. Jr. Nucl. Sci. Eng.; 64: No. 2, 418-451 (Oct 1977).

The paper reviews major computer code packages presently utilized or under active development in the U.S. for deterministic liquid-metal fast breeder reactor safety analysis. Although the review covers the very large accident domain from intact predisassembly geometry through radiological consequences, an attempt was made to categorize code systems according to the physical problems solved to provide a working orientation for those not intimately involved in the field. Mathematical and computational aspects were given particular focus in the hope of stimulating development of more efficient numerical approaches in those areas of greatest need.

- 4335 PROBABILISTIC FAST REACTOR ACCIDENT ANALYSIS. Ott, K.O. (Purdue Univ., West Lafayette, IN). Nucl. Sci. Eng.; 64: No. 2, 452-464 (Oct 1977).

An approach for the description and evaluation of the uncertainties in the predicted consequences of fast breeder reactor accidents is outlined. All uncertainties are viewed as uncertainties of parameters, either of physical parameters or of quantities that appear in the parameterization of phenomenological models. By the incorporation of parameter uncertainties in the simulation of the accident progression, single accident-path scenarios assume the character of "accident spectra." The progression of accident spectra is found by combining deterministically calculated accident-path results with the probability of the respective set of input parameters. The substantial method development needed for the implementation of the approach is discussed, and the status of the development is briefly reviewed. Typical results are presented for illustration purposes. The possible eventual significance of the approach is indicated.

- 4336 NUMERICAL MODEL OF REACTOR FUEL AND COOLANT MOTIONS FOLLOWING PIN FAILURE. Pizzica, P.A.; Abramson, P.B. (Argonne National Lab., IL). Nucl. Sci. Eng.; 64: No. 2, 465-479 (Oct 1977).

From Proceedings of the international meeting on fast reactor safety and related physics; Chicago, IL, USA (Oct 1976).

The computer code EPIC models fuel and coolant motion that results from internal fuel pin pressure (from fission gas or fuel vapor) and/or from the generation of sodium vapor pressures in the coolant channel subsequent to pin failure in a liquid-metal fast breeder reactor. The modeling includes the ejection of molten fuel from the pin into a coolant channel with any amount of voiding through a clad rip which may be of any length or which may expand with time. One-dimensional Eulerian hydrodynamics is used to model both the motion of fuel and fission gas inside a molten fuel cavity and the mixture of two-phase sodium and fission gas in the channel. Motion of molten fuel particles in the coolant channel is tracked with a particle-in-cell technique.

- 4337 SIMULATION OF TRANSIENTS IN LIQUID-METAL FAST BREEDER REACTOR SYSTEMS. Agrawal, A.K.; Guppy, J.G.; Madni, I.K.; Quan, V.; Weaver, W.L. III; Yang, J.W. (Brookhaven National Lab., Upton, NY). Nucl. Sci. Eng.; 64: No.

2, 480-491 (Oct 1977).

The transient simulation of a liquid-metal fast breeder reactor (LMFBR) plant requires (a) modeling of all processes that may be encountered and (b) the development of numerical methods to solve them. All models needed for the thermohydraulic simulation of the whole plant are formulated. Numerical techniques required to solve the governing equations, which are hyperbolic and parabolic partial-differential equations and ordinary differential equations, are examined. It appears that the implicit (or partially implicit) scheme is most suitable to meet both the stability and accuracy requirements. A new approach, labeled as the multistep scheme, to efficiently solve the entire system is then presented and illustrated through an example. For a simplified test problem, the multistep scheme has been found to be more efficient (by a factor of 2 to 3) than the commonly used single-step methods. This effort has resulted in the creation of a system transient simulation code, called SSC, for LMFBRs.

- 4338 (ANL-Trans--1108) DEVICE FOR COOLING THE UPPER PART OF THE VESSEL OF A LIQUID SODIUM COOLED REACTOR. Gama, J.M.; Lallement, J.; Pugnot, L. Oct 1977. Translation of French Patent application No. 7,337,232 13p. Dep. NTIS, PC A02/MF A01.

The invention deals with a device for cooling the main vessel of a nuclear reactor in which the primary cooling circuit is located completely within the main reactor vessel.

- 4339 (ANL-Trans--1109) DEVICE FOR CONNECTING TWO TUBULAR DUCTS MOUNTED END-TO-END. Bentz, A.; Deplierre, Y.; Venobre, H. Oct 1977. Translation of French Patent application No. 7,331,539 12p. Dep. NTIS, PC A02/MF A01.

The invention concerns a device for connecting two tubular ducts, particularly two pipelines traversed by a liquid under pressure, this device allowing a relative displacement of these ducts in any direction, either along the direction of their common axis or in a plane perpendicular to this direction, or in any direction combining one and the other of these two displacements, while assuring free flow of the liquid from one duct to the other. The invention applies particularly, although not exclusively, when the ducts consist of pipelines installed in the interior of a sodium-cooled fast nuclear reactor, the device assuring connection of the outlet diffuser duct of a primary pump forming the sodium circulation in the tank through the core and heat exchangers, and of a large-diameter pipeline conducting the sodium thus delivered by the pump under the core support slab.

- 4340 US BREEDER REACTOR PROGRAMME. Beckjord, E.S. (Energy Research and Development Administration, Washington, D.C. (USA). Div. of Reactor Development and Demonstration). J. Br. Nucl. Energy Soc.; 16: No. 4, 323-326 (Oct 1977).

Lecture given at the joint meeting of the Risley Nuclear Engineering Society.

Changes in the US breeder reactor programme, as a result of recent nuclear power policy guidelines, are discussed. The activities and recommendations of the Steering Committee established by ERDA for its review of LMFBRs and the Clinch River Breeder Reactor, in particular, are described. Based on majority and minority reports of its members, the summary report of the Committee provided input for the National Energy Plan announced to

Congress in April 1977. Many of the plans new proposals are based on concern over proliferation and especially plutonium-based fuel cycles. The effect of the proposed policies both on LWRs and on the LMFBR programme are examined. Finally an outline is given of the objectives and probable future activities relating to fast breeder reactors in the U.S.A.

- 4341 **STRESS EFFECT ON THE VOID SWELLING INCUBATION PERIOD.** Weiner, R.A.; Boltax, A. (Westinghouse Electric Corp., Madison, Pa. (USA). Advanced Reactors Div.). J. Nucl. Mater.; 68: No. 2, 141-147 (Oct 1977).

The dose dependence of the swelling of solution-annealed Type 316 stainless steel irradiated as fuel pin cladding is compared with the behavior of irradiated stress-free samples. It is found that the onset of steady-state swelling occurs at a considerably lower fluence level for fuel pin cladding and a stress effect on the swelling incubation period is postulated to explain this result. A constitutive equation for this stress effect is formulated and a technique for using this constitutive equation with time-dependent stress histories is described. The stress effect on the incubation period is used in the calculation of the axial swelling profile of a fuel pin with 20% cold-worked Type 316 stainless steel cladding and good agreement with the measured swelling profile is obtained.

- 4342 **SODIUM TEST RIGS FOR BRITAIN'S COMMERCIAL REACTOR.** Masters, R. Nucl. Eng. Int.; 22: No. 263, 33-34 (Oct 1977).

Two new sodium rigs which are undergoing dry commissioning at Nuclear Power Development Laboratories, Risley, are described. The high temperature sodium loop and the sodium components test rig are designed for testing, in sodium, large scale models and components for use in the Commercial Fast Reactor. Workshop and control facilities are also described.

- 4343 **EQUIPMENT FOR THE NONDESTRUCTIVE TESTING OF 2-1/4 CR-1 MO DUPLEX LMFBR STEAM GENERATOR TUBING.** Longua, K.J.; Whitham, G.K.; McElroy, J.T. (Argonne Natl Lab, Idaho Falls). Mater. Eval.; 35: No. 10, 54-59 (Oct 1977).

The development of a nondestructive testing method was required for examination of the duplex tubing in the evaporator of the steam generator in the Experimental Breeder Reactor No. II (EBR-II). Ultrasonic testing was the only viable means of testing the steam generator tubing because the tubing material (2 1/4% Cr, 1 Mo) is a ferritic steel. The equipment designed to perform this inspection consisted of an ultrasonic inspection probe, a drive unit to rotate and move the probe, and a standard that contained electrical-discharged-machined (EDM) notches to be used to estimate flaw depth. The equipment is capable of inspecting either single wall or bonded duplex tubing with relative ease and reasonably fast. 1 ref.

- 4344 **WHAT WILL THE FAST BREEDER BRING US. AS AN IMPORTANT INTERLUDE, NUCLEAR POWER UTILIZATION MAY BE PROLONGED BY FAST BREEDERS: BUT IT MAY ALSO END AS A CATASTROPHE.** Urban, M. Sulzer Tech. Rev.; No. 226, 8 (Oct 1977). (In German).

This newspaper article gives a survey of the problems in connection with the fast breeder which are now under public discussion.

- 4345 **DECISION ON 'FAST BREEDER' PASSED ON TO FEDERAL CONSTITUTIONAL COURT.** Buedenbender, U. Energiewirtsch. Tagesfragen; 27: No. 10, 705-712 (Oct 1977). (In German).

The decision of the Higher Administrative Court at Muenster dated 18th August 1977 - VII A 338/74 - has been passed on to those concerned. The wording is as follows: 'The proceedings are suspended. A decision by the Federal Constitutional Court will be obtained as to whether section 7 of the Act on the peaceful uses of nuclear energy and the protection against its hazards of December 23rd, 1959, BGBI I 814, in the version of the publication of October 31st, 1976, BGBI I 3053, as far as it enables the licensing of nuclear power plants of the fast breeder type, is within the Basic Law.' The main grounds upon which the decision is based with comments are given.

- 4346 **FAST REACTOR INSTRUMENTATION DEVELOPMENT IS AIMED AT A COMMERCIAL PLANT.** Hughes, G.; Hayes, D.J.; Hobdell, M.R. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). Nucl. Eng. Int.; 22: No. 263, 45-50 (Oct 1977).

Problems and present work relating to the provision of suitable instrumentation for commercial fast reactors are reviewed. The areas involved are: safety instrumentation, which includes development for improving the sensitivity and speed of response of sensors for measurement of temperature, flow, and pressure, and detection for boilers and for sodium systems; meters for monitoring impurities in sodium and in cover gas regions; and finally monitoring and inspection of reactor components.

- 4347 **REVIEW OF THE POTENTIAL USE OF NUCLEAR REACTORS FOR HYDROGEN PRODUCTION.** Worley, N.G. (Babcock and Wilcox (Ops) Ltd, London, Engl.). Gas Eng. Manage.; 17: No. 10, 371-383 (Oct 1977).

Except for marine propulsion, nuclear power, up to now, has only been used for electricity generation. The paper reviews the types of nuclear reactors available, the characteristics of the electricity industry and the way that nuclear power could supplement the supplies of gas when these become less abundant early in the twenty-first century. The route using high temperature reactors appears basically less attractive than electrolysis particularly when the increasing need for nuclear reactors which can 'breed' new fuel is taken into consideration. Some general comments on electrolysis plant are made together with some indications of the type of problem involved in their integration into the electricity/gas structure. 8 refs.

- 4348 **CHARACTERISTICS OF THE COMBINED OPERATION OF THERMAL AND FAST POWER REACTORS.** Litkin, V.B.; Kagramanyan, V.S. J. Br. Nucl. Energy Soc.; 16: No. 4, 335-339 (Oct 1977).

From Operating experience with nuclear power stations and routes of future development of atomic power conference; Obninsk, USSR (25 - 27 Jun 1974).

Thermal reactors have several deficiencies, in particular, their low efficiency in the use of national uranium resources. The advantages of joint use of thermal and fast reactors in a nuclear power system are indicated. Current fast reactors assessments however, show higher specific capital costs than for thermal reactors. To determine the combination of both types for which the economic effect is a

maximum, optimization calculations are necessary including optimization of reactor characteristics. Some of the problems involved are examined. Analysis of the results shows that in the period of mass introduction of fast reactors, thermal reactors of high plutonium production are required in the early stages and that the significance of expenditure on the initial fuel inventory in the cycle increases sharply in comparison with that for established fast reactors.

- 4349 NUCLEAR FUEL CYCLE. PT. 4. THE REACTORS. Farmer, A.A. (UKAEA Reactor Group, Risley). Arch. Energiewirtschaft.; 31: No. 10, 843-858(Oct 1977). (In German).

The basic concepts of the reactor types now in commercial operation or to be realized before 1990 are described in the form of a survey. In the framework of a description of special features and differences, the typical parameters of the reactors are listed for comparison.

- 4350 PERFORMANCE OF AN ELECTROMAGNETIC TEMPERATURE SENSOR IN LIQUID SODIUM. Farrington, B.J.; Hughes, G. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). J. Br. Nucl. Energy Soc.; 16: No. 4, 347-352(Oct 1977).

An electromagnetic temperature sensor has been evaluated in flowing sodium. Temperature fluctuations, produced by the injection of a hot stream of sodium into the main sodium flow, have been measured with the sensor and compared with the indications of a small-diameter thermocouple. These comparisons show that a directly immersed sensor is capable of detecting 0.1 degC root-mean-square temperature oscillations with a 20 db signal-to-noise ratio, and enable the temperature sensitivity of the sensor to be determined. The sensor has also been operated from within a stainless steel pocket at reduced sensitivity (approximately 40% of the directly immersed value). In this operational mode the measured sensitivity agrees closely with that predicted by a developed theoretical analysis technique.

- 4351 SAFETY AND RELIABILITY: ANALYSES OF A BREEDER REACTOR. Fussell, J.B.; Cate, C.L.; Wagner, D.P. (Univ. of Tennessee, Knoxville). Simulation; 29: No. 4, 121-124(Oct 1977).

A University of Tennessee independent study of the safety and reliability of the Clinch River Breeder Reactor Plant in Oak Ridge, Tennessee is described. The study uses several kinds of models, and analytical computer programs, one of which (BACFIRE) is briefly described and illustrated.

- 4352 INVESTIGATION OF MEASURING ACCURACY OF PLUGGING INDICATORS. Yamamoto, H.; Murase, M.; Izumi, S.; Sagawa, N. (Hitachi Ltd, At Energy Res Lab, Ozenji, Kawasaki-shi, Jpn). J. Nucl. Sci. Technol. (Tokyo); 14: No. 10, 689-694(Oct 1977).

In order to improve the measuring accuracy of plugging indicators, the effects of operating conditions such as the sodium flow rate and the cooling air flow rate on the measuring error are analyzed. By this investigation, the measuring error can be evaluated as a function of the Reynolds and Stanton numbers, and suitable conditions which minimize the measuring error are obtained. 6 refs.

- 4353 SWORD OF DAMOCLES, OR: THE FAST BREEDER

IN KALKAR AND THE HIGHER ADMINISTRATIVE COURT AT MUENSTER. Dreistern. Energie; 29: No. 10, 307-309(Oct 1977). (In German).

A decision of the OVG Muenster of August 18th, 1977 has made the headlines. The judges of Muenster have put the question to the Federal Administrative Court, whether section 7 of the Atomic Energy Act is in agreement with the constitution as far as the licensing of fast breeders is concerned. A technician comments on the explanations of the OVG judges in a press conference as regards their decision.

- 4354 STRAIN RATE EFFECTS ON THE ELEVATED-TEMPERATURE TENSILE BEHAVIOR OF A BAINITIC 2 1/4 CR-1 MO STEEL. Klueh, R.L. (Oak Ridge National Lab., TN); Oakes, R.E. Jr. J. Eng. Mater. Technol.; 99: No. 4, 350-358(Oct 1977).

The tensile properties of a normalized-and-tempered 2-1/4 Cr-1 Mo steel were determined from 25 to 566°C and the strain rate 2.67 x 10⁻⁴ to 144/s. The specimens were taken from a 1-in. thick plate and had a microstructure that was essentially 100 percent bainite. Except at 25 and 566°C, the 0.2 percent yield stress was little affected by strain rate; at 25 and 566°C, the yield stress increased with increasing strain rate. At a constant strain rate, the yield stress decreased with increasing temperature. The effect of strain rate and temperature on the ultimate tensile strength was more complicated. A strength peak that resulted from dynamic strain aging was observed in the ultimate tensile strength-temperature relationship. The position of these peaks moved to higher temperatures with increasing strain rate. Total elongation and reduction of area were relatively constant over the range of test variables, except at 566°C, where they increased with decreasing strain rate. However, uniform elongation decreased with decreasing strain rate at 510 and 566°C, dropping to 1 and 0.6 percent, respectively.

- 4355 REPRESENTATION OF ELEVATED-TEMPERATURE TENSILE BEHAVIOR OF TYPE 304 STAINLESS STEEL IN A SODIUM ENVIRONMENT. Chopra, O.K.; Natesan, K. (Argonne National Lab., IL). J. Eng. Mater. Technol.; 99: No. 4, 366-371(Oct 1977).

An analysis of tensile behavior of Type 304 stainless steel was conducted for specimens in the solution-annealed condition and after exposure to a sodium environment. The Voce equation was used to describe tensile flow curves for plastic strains above 0.005 at temperatures between 550 and 700°C and strain rates of 3.81 x 10⁻⁶ to 1.90 x 10⁻³ s⁻¹. The results show that, when compared with solution-annealed specimens, the tensile flow behavior of the sodium-exposed specimens is characterized by a higher strain-hardening rate, which decreases rapidly with an increase in flow stress. The values of the saturation stress for uniform elongation predicted from the Voce model are higher for the sodium-exposed specimens than for those in the solution-annealed condition at strain rates < or approximately 5 x 10⁻⁵ s⁻¹ and lower for strain rates > or approximately 5 x 10⁻⁵ s⁻¹. Metallographic examination of the fracture surfaces shows a transition from a complete ductile fracture to a partial intergranular failure as the strain rate decreases. Carburization of the specimens appears to inhibit the intergranular failure.

- 4356 EXPERIMENTAL STUDIES IN EXTINGUISHING LARGE SODIUM FIRES. Vylomov, V.V.; Ivanenko, V.N.; Ivanov, B.G.; Karpov, A.V.; Kirichenko, V.N.; Kozlov, V.I.; Krutov, V.A.; Mikhedov, V.G. At. Energ. (USSR); 43: No. 4, 286-

- 288(Oct 1977). (In Russian).
TL
- 4357 DETERMINATION OF CESIUM CONTENT IN THE BOR-60 PRIMARY COOLING CIRCUIT. Poznyak, G.I.; Privalova, P.A.; Timofeev, G.A. At. Energ. (USSR); 43: No. 4, 288(Oct 1977). (In Russian).
BOR-60 REACTOR; CESIUM 133; CESIUM 135; CESIUM 137; ISOTOPE RATIO; MASS SPECTROSCOPY; PRIMARY COOLANT CIRCUITS; SAMPLE PREPARATION; SODIUM; RADIOACTIVITY TRANSPORT
- 4358 STUDIES OF THERMAL-HYDRODYNAMIC FLOW INSTABILITY, 2. EFFECTS OF HEATING CONDITION OF THERMAL-HYDRODYNAMIC FLOW INSTABILITY. Suzuki, A. (Osaka Univ. (Japan)). Bull. JSME (Jpn. Soc. Mech. Eng.); 20: No. 148, 1291-1298(Oct 1977).
For reliable prediction of flow stability in sodium-heated steam generators, a dynamic model was proposed for boiling flow oscillation in parallel channel systems, and an analysis code was developed. The model contains a description of a sodium flow exchanging heat with a water flow in counter-current fashion. The code was applied to three representative flow systems whose heating conditions differed from each other, whereby their flow stabilities were compared with a focus on the effects of heating condition. Eigenvalues and flow impedances of the oscillation determined for each system reveal that: (1) Two fundamental systems for the steam generator, parallel tube system in an evaporator and steam generator modules arranged in parallel, have different stabilities under low frequency oscillation. (2) Existing analysis model conditioned on constant heat flux gives different results on stability from those of either steam generator model under low frequency oscillation.
- 4359 THEORETICAL MODEL OF AN EDDY-CURRENT FLOWSENSOR. Hirayama, M. (Toshiba Res and Dev Cent, Ukishimacho, Kanagawa, Jpn). IEEE Trans. Nucl. Sci.; NS-24: No. 5, 2021-2030(Oct 1977).
A theoretical model of an eddy-current flowsensor has been derived to determine its optimum design for sensitivity. The validity of the model has been confirmed by experiments using a moving aluminum cylinder as a fluid. 8 refs.
- 4360 ACCIDENT ALARM IN STEAM GENERATORS OF SODIUM COOLED FAST REACTOR POWER PLANTS. I. TESTING THE FUNCTION OF AN ION PUMP PH-METER ON SIMULATED WATER PENETRATION IN LIQUID SODIUM. Pitak, O. (Chemoprojekt, Prague (Czechoslovakia)); Fresl, M. (Statni Vyzkumny Ustav Ochrany Materialu G.V. Akimova, Prague (Czechoslovakia)); Matal, O.; Martoch, J. (Vyzkumny Ustav Zavodu Energetického Strojirenstvi, Brno-Slatina (Czechoslovakia)). Jad. Energ.; 23: No. 10, 369-372(Oct 1977). (In Czech).
An experimental loop was used for modelling water penetration into liquid sodium. The experiment aimed at testing the suitability of a pH-meter using the principle of hydrogen diffusion in sodium. Sodium was pumped by an ion pump. Hydrogen concentration in sodium was considerably high; sodium was hydrogen-saturated. It was shown that water penetration could also be measured at high hydrogen concentration albeit with reduced sensitivity. The maximum pH-meter sensitivity under the given conditions corresponded to 5% change in the initial concentration. The response rate of the pH-meter with a probe having a membrane wall thickness of 2 mm and effective area of 130 cm² ranged between 30 and 45 s.
- 4361 PROCESS AND DEVICE FOR THERMAL ENERGY PRODUCTION. Mangus, J.D. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). French Patent 2,366,444/A/. 29 Sep 1977. Priority date 30 Sep 1976, United States of America (USA), 15p. (In French).
The main aim of the invention is to create a heating cycle arrangement, for the energy production facilities as from liquid metal cooled nuclear reactors, that will stand up to the temperature changes of the heated steam at least as from the high pressure turbine. This arrangement includes a first system in which flows a liquid metal coolant between a heat source, a steam generator and a utilisation system on which flows a vaporisable fluid from this generator, passing through a first turbine, a heater, at least a second turbine and a condenser. The steam heated in the heater is heated by the liquid metal coolant. A preheater is located in the heated steam system upstream of the heater. This preheater is connected so as to heat the steam to a preset, practically constant value, before this steam to be heated enters the heater heated by the liquid metal. This arrangement reduces the thermal transitions in the superheater and the heater during load changes. In a preferential design mode, the steam from the steam generator is sent to a moisture extraction drum and the heater is exposed to the steam in this drum.
- 4362 HEAT-INSULATED DEVICE, FOR INSULATING THE TOP PART OF THE ANNULAR SPACE BETWEEN THE MAIN VESSEL AND THE SAFETY VESSEL OF A FAST-NEUTRON REACTOR. Lemerrier, G. (to Commissariat a l'Energie Atomique). US Patent 4,050,988. 27 Sep 1977. Priority date 30 Aug 1974, France, 4p.
The invention relates to a device for heat-insulating the top part of the annular space separating the main vessel from the safety vessel from a fast-neutron nuclear reactor, the two vessels having a common vertical axis and being open at their top part and covered by a horizontal slab in which the ends are fitted so as to be suspended by the slab. According to the invention a substantially cylindrical heat-insulating ring is mounted under the slab and cooperates with the slab and with the facing walls of the vessel to bound the region to be insulated. The ring extends through the annular space in a horizontal plane and is carried by the slab by means of tie-rods, and a top insulating ring divided into two coaxial elements rests on the bottom ring and is secured to the walls of the two vessels respectively by an assembly of spacing springs disposed radially and bearing against the two elements. 8 claims, 3 figures.
- 4363 NUCLEAR REACTOR. Yant, H.W.; Stinebiser, K.W.; Anzur, G.C. (to Westinghouse Electric Corp.). US Patent 4,050,985. 27 Sep 1977. Filed date 13 Sep 1974. 16p.
Disclosed is a liquid-metal breeder reactor whose upper internals include outlet modules for channeling the liquid-metal coolant from selected areas of the outlet of the core vertically to the outlet plenum. The modules are composed of a highly-refractory, high corrosion-resistant alloy, for example, INCO-NEL-718. Each module is disposed to confine and channel generally vertically the coolant emitted from a subplurality of core-component assemblies. Each module has a grid with openings, each opening disposed to receive the

coolant from an assembly of the subplurality. The grid in addition serves as a holdown for the assemblies of the corresponding subplurality preventing their excessive ejection upwardly from the core. In the region directly over the core the outlet modules are of such peripheral form that they nest forming a continuum over the core-component assemblies whose outlet coolant they confine. Each subassembly includes a chimney which confines the coolant emitted by its corresponding subassemblies to generally vertical flow between the outlet of the core and the outlet plenum. Each subplurality of assemblies whose emitted coolant is confined by an outlet module includes assemblies which emit lower-temperature coolant, for example, a control-rod assembly, or fertile assemblies, and assemblies which emit coolant of substantially higher temperature, for example, fuel-rod assemblies. The coolants of different temperatures are mixed in the chimneys reducing the effect of stripping (hot-cold temperature fluctuations) on the remainder of the upper internals which are composed typically of AISI-304 or AISI-316 stainless steel. 7 claims, 28 figures.

- 4364 ROTATING PLUG BEARING AND SEAL. Wade, E.E. (to Westinghouse Electric Corp.). US Patent 4,050,987. 27 Sep 1977. Filed date 30 Dec 1974. 6p.

Disclosed is a bearing and seal structure for nuclear reactors utilizing rotating plugs above the nuclear reactor vessel. The structure permits lubrication of bearings and seals of the rotating plugs without risk of the lubricant draining into the reactor vessel below. The structure permits lubrication by utilizing a rotating outer race bearing. 19 claims, 3 figures.

- 4365 (BNWL-tr--265) RESULTS ON AN IRRADIATION EXPERIMENT ON ABSORBER MATERIALS FOR FAST SODIUM-COOLED REACTORS. Hoerle, M.; Enderlein, H.; Pejisa, R.; Duennen, P. (Battelle Pacific Northwest Labs., Richland, WA (USA)). 20 Sep 1977. Translation of AED-Conf--1976-013-195 (CONF-760359--6). 8p. Dep. NTIS, PC A02/MF A01.

From Reactor meeting; Duesseldorf, F.R. Germany (30 Mar 1976).

Some results concerning helium release, dimensional stability, swelling, density change, and cladding compatibility of irradiated LMFBR absorber materials are presented. The materials included B_4C , Eu_2O_3 , and EuB_6 .

- 4366 STRUCTURAL DESIGN OPTIMIZATION OF ONCE-THROUGH TYPE HEAT EXCHANGERS. Chiang, C.C. (Combust Eng, Inc, Windsor, Conn). Am. Soc. Mech. Eng., [Pap.]; No. 77-JPGC-NE-3, vp(18 Sep 1977).

This paper provides a general solution for the structural design of once-through type heat exchangers. From this solution, detailed equations are developed for optimizing the design of once-through units. In the analysis, use of the functions of the stiffness of the shell expansion joint and the size factor of the unperforated annular ring of the tube sheet greatly increases the design flexibility and provides a more direct solution for the structural design of once-through type heat exchangers. Two sample units, one a fixed-shell model and the other a floating-shell model, are analyzed to demonstrate that by using the proposed method an optimum design can be achieved. 4 refs.

- 4367 PROGRAM TO ESTABLISH LMFBR COMMERCIALIZATION. Richards, R.B.; Palmer, R.S. (GE, Sunnyvale, Calif). Am. Soc. Mech. Eng., [Pap.]; No. 77-JPGC-NE-23, vp(18 Sep 1977).

Some of the lessons of LWR commercialization are identified and examined; from these, a unique program for the commercialization of the LMFBR is proposed and a schedule for its accomplishment is presented and discussed. The proposed program features a standardized plant design reducing development effort while promoting reliable operation, high plant availability, and a standard licensing procedure. The result of this program would be increased utility confidence in the LMFBR, and greater public acceptance of this important energy source.

- 4368 CORROSION TESTING UNDER DEPARTURE FROM NUCLEATE BOILING CONDITIONS MODELING THE CLINCH RIVER BREEDER REACTOR PLANT STEAM GENERATOR. Rodgers, D.N.; Dutina, D.; Rush, D.J.; Stephens, P.B.; Valente, C.G. (GE, San Jose, Calif). Am. Soc. Mech. Eng., [Pap.]; No. 77-JPGC-NE-8, vp(18 Sep 1977).

Evaporators of a CRBR plant experience departure from nucleate boiling in the tubes under normal power conditions. Potential damage mechanisms associated with DNB include thermal strain cycling which may lead to exfoliation of the protective oxide layer on the steam side of the evaporator tube and water impurity concentration at DNB which may lead to increased local corrosion. After 2820 hr of testing under expected "worst case" thermal and water chemistry conditions in a sodium-heated single-tube model evaporator, non-destructive examination shows that the postulated accelerated corrosion resulting from these mechanisms has not occurred. The test loop, test model, and test operation designed to simulate the thermal hydraulic and water chemistry conditions expected for the plant evaporators and establish nucleate boiling and DNB regimes in the test model are described. 5 refs.

- 4369 EXPERIMENTAL EVALUATION OF DUPLEX TUBE HEAT TRANSFER CHARACTERISTICS SIMULATING STEAM GENERATOR EXTERNAL HEATING. Efferding, L.E.; Hwang, J.Y. (Westinghouse Electr Corp, Tampa, Fla). Am. Soc. Mech. Eng., [Pap.]; No. 77-JPGC-NE-9, vp(18 Sep 1977).

An experimental heat transfer evaluation of archived EBR-II 2-1/4Cr-1Mo steam generator double-wall (duplex) tubes in both the as-fabricated prestressed and annealed conditions has been completed which has resulted in the determination of thermal performance of the interface clearance behavior under heating conditions simulating that of a sodium-heated steam generator. The conclusions drawn from the EBR-II tube evaluation contribute to the establishment of design criteria and analyses techniques for duplex tubes in a steam generator application. Tests performed on various sample tubes reflected annealed and prestressed conditions with clean, oxygen, and moisture-contaminated interfaces and various initial interface clearances. Unique testing techniques and data analysis evaluation methods were devised which results in a direct resolution of the interface thermal resistance behavior. 13 refs.

- 4370 INVESTIGATIONS INTO THE SODIUM-STEAM SURFACE INTERACTION ON A STAGNANT POOL. Stehling, S.G.; Hunsicker, J.C. (GE, Sunnyvale, Calif). Am. Soc. Mech. Eng., [Pap.]; No. 77-JPGC-NE-11, vp(18 Sep 1977).

Experimental investigations into the interface reaction between a steam atmosphere and a stagnant sodium pool have been conducted at the General Electric San Jose Test Facility. The objectives of the tests were to create worst case accident situations and evaluate the consequences with respect to the design of sodium heated steam generators. The most important results of these investigations were the creation and actual photographing of a "pool fire" and the determination that damage mechanism associated with it is of a corrosive nature rather than thermal weakening as previously suspected. 1 ref.

- 4371 OPERATING EXPERIENCE AND SODIUM-WATER REACTION TESTING IN THE LARGE LEAK TEST RIG. DeBear, W.S.; Neely, H.H. (Liq Met Eng Cent, Canoga Park, Calif). Am. Soc. Mech. Eng., [Pap.]; No. 77-JPGC-NE-6, vp(18 Sep 1977).

The Large Leak Test Rig (LLTR) and its role in providing information on the consequences of large sodium-water reactions in sodium-heated steam generators is described. Facility startup problems and subsequent operating experiences are discussed. Instrumentation and data acquisition methods are identified and typical data from initial tests using the 30-MWt Modular Steam Generator (hockey-stick steam generator) as a test article are presented. Information also is included on post-test inspection methods, equipment, and results. 4 refs.

- 4372 DEVELOPMENT TESTING OF LEAK INJECTION DEVICES FOR USE IN TESTING SODIUM-WATER LEAK DETECTION MODULES. Stehling, S.G.; Hunsicker, J.C.; Sperberg, R.T. (GE, Sunnyvale, Calif). Am. Soc. Mech. Eng., [Pap.]; No. 77-JPGC-NE-7, vp(18 Sep 1977).

A leak detection device has been designed and tested. Small quantities of water or hydrogen are injected into a steam generator in order to test the sensitivity and calibration of the leak detectors. The outstanding feature of this device is that it may be mounted inside the tube bundle of a steam generator, because water injections made with the device produce negligible wastage of the tubes. This report describes the tests utilized to verify the design of leak injectors to be installed in the Few Tube Test Model, located in General Electric's Steam Test Rig, San Jose, California. The most significant result of this program was that it was demonstrated that a safe, dependable design for a Leak Injection Device has been achieved. 3 refs.

- 4373 FAST NUCLEAR REACTOR. Kovin, V.M.; Mitenkov, F.M.; Ovechkin, A.V. US Patent 4,048,011. 13 Sep 1977. Filed date 13 Sep 1974. 6p.

The proposed fast nuclear reactor comprises a vessel, a liquid metal coolant filling said vessel, the side wall of which is cylindrical in shape and has its axis disposed in a horizontal plane. The vessel accommodates a core and heat exchangers with said liquid-metal coolant being passed therethrough by means of forced circulation thereof. The space above the coolant level is occupied by an inert gaseous medium, with the coolant contacting said medium in pipe branches penetrating the side wall of the vessel. 5 claims, 4 figures.

- 4374 WALL PENETRATION OF A PIPE OPERATED AT HIGH TEMPERATURES, ESPECIALLY LINER PENETRATION. Kappauf, H. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). German (FRG) Patent

2,609,088/A/. 8 Sep 1977. 9p. (In German).

In order to counter the difficulties occurring due to thermal stresses when a pipe in which a hot medium (e.g. sodium at 500° C) is flowing is lead through a wall it is proposed to introduce a spring element between wall and pipe. This spring element absorbs the positional changes and prevents the build-up of high stresses. Details of the design are described.

- 4375 EDDY CURRENT POSITION INDICATING APPARATUS FOR MEASURING DISPLACEMENTS OF CORE COMPONENTS OF A LIQUID METAL NUCLEAR REACTOR. Day, C.K.; Stringer, J.L. (to Westinghouse Electric Corp.). US Patent 4,047,103. 6 Sep 1977. Filed date 4 Feb 1975. 6p.

Apparatus for measuring displacements of core components of a liquid metal fast breeder reactor by means of an eddy current probe is described. The active portion of the probe is located within a dry thimble which is supported on a stationary portion of the reactor core support structure. Split rings of metal, having a resistivity significantly different than sodium, are fixedly mounted on the core component to be monitored. The split rings are slidably positioned around, concentric with the probe and symmetrically situated along the axis of the probe so that motion of the ring along the axis of the probe produces a proportional change in the probe's electrical output. 11 claims, 3 figures.

- 4376 NUCLEAR REACTORS. Durston, J.G.; Hind, J.R. (to United Kingdom Atomic Energy Authority). US Patent 4,046,629. 6 Sep 1977. Priority date 7 Jul 1975, United Kingdom of Great Britain and Northern Ireland (UK), 4p.

Thermal insulation for a core tank of a fast breeder nuclear reactor comprises a plurality of spaced membranes defining a series of concentric chambers lining the tank. Liquid metal is pumped from the cooler regions of a pool of coolant into the first chamber from which it flows successively through the remaining chambers towards the reactor core. 2 claims, 3 figures.

- 4377 PIPE SUPPORT FOR USE IN A NUCLEAR SYSTEM. Pollono, L.P.; Mello, R.M. (to Energy Research and Development Administration). US Patent 4,046,169. 6 Sep 1977. Filed date 9 Jul 1975. 10p. PAT-APPL-594,477.

Disclosed is a pipe support for high temperature, thin-walled vertical piping runs used in a nuclear system. A cylindrical pipe transition member, having the same inside diameter as the thin-walled piping, replaces a portion of the piping where support is desired. The outside diameter of the pipe transition member varies axially along its vertical dimension. For a section of the axial length adjacent the upper and lower terminations of the pipe transition member, the outside diameter is the same as the outside diameter of the thin-walled piping to which it is affixed. Intermediate of the termination sections, the outside diameter increases from the top of the member to the bottom. Adjacent the lower termination section, the diameter abruptly becomes the same as the piping. Thus, the cylindrical transition member is formed to have a generally triangular shaped cross-section along the axial dimension. Load-bearing insulation is installed next to the periphery of the member and is kept in place by an outer ring clamp. The outer ring clamp is connected

to pipe hangers, which provide the desired support for the vertical thin-walled piping runs.

- 4378 WHY THE BREEDER REACTOR IS INEVITABLE. Alexander, T. Fortune; 122-130(Sep 1977).

The introduction of the liquid metal fast breeder reactor in the United States is necessary if nuclear energy is to supply a significant portion of future energy needs. The political, technical, and management problems associated with breeder development in the U.S. are reviewed. Political and management aspects are seen as the major obstacles to overcome as the technical problems have been largely solved in both domestic and foreign development programs.

- 4379 BENCHMARK CALCULATIONS FOR A SODIUM-COOLED BREEDER REACTOR BY TWO- AND THREE-DIMENSIONAL DIFFUSION METHODS. Buckel, G.; Kuefner, K.; Stehle, B. (Kernforschungszentrum Karlsruhe, Ger.). Nucl. Sci. Eng.; 64: No. 1, 75-89(Sep 1977).

This investigation concentrates on the numerical solution of the multigroup neutron diffusion equations by computer codes. For a realistic model liquid-metal fast breeder reactor, several benchmark problems in two and three space dimensions were derived and calculations were performed by eight different computer programs. The effect on k_{eff} and the neutron fluxes of the refinement of the discretization mesh is studied. Very good agreement (approximately 0.05%) of the results was found in those cases where the computer programs use the same discretization scheme of mesh-edged discretization formulas, although the codes employ different methods of solution. On the other hand, minor discrepancies remain between results obtained by codes using mesh-edged and mesh-centered discretization formulas, even for fine-mesh grids. The reasons are not understood in every detail. Fortunately, these discrepancies are very small and more of theoretical than practical interest. The effect of a simple group condensation scheme of k_{eff} was also investigated by considering several different energy group structures. Spatial mesh refinements and resolution of the energy range were found to be well decoupled. As the main result, one may take the fact that spatial and energetic mesh refinements may influence the results rather strongly, unless the mesh step is comparable to the minimum diffusion length and unless enough energy groups are used.

- 4380 BENCHMARK ANALYSIS OF LIQUID-METAL FAST BREEDER REACTOR NUCLEAR DESIGN METHODS. Kujawski, E.; Bailey, H.S. (General Electric Co., Sunnyvale, CA). Nucl. Sci. Eng.; 64: No. 1, 90-94(Sep 1977).

The Large Core Code Evaluation Working Group has been organized for the prime aim of testing and validating neutronics codes and methods for use in the analysis of large fast reactors. For the initial problem, static neutronics calculations were performed on a "representative" 1200-MW(e) liquid-metal fast breeder reactor in a two-dimensional radial geometry. The initial results representing the cooperative effort of the participants are summarized. Various codes and methods using few-group diffusion theory are intercompared. The calculational methods are in substantial agreement provided the same multigroup constants are used. This effort is to be further pursued and broadened to investigate such specific effects as sodium voiding.

- 4381 THREE-DIMENSIONAL THERMAL-HYDRAULIC ANALYSIS OF WIRE-WAPPED RODS IN LIQUID-METAL FAST BREEDER REACTOR CORE ASSEMBLIES. Chuang, M.C. (Westinghouse Research Labs., Pittsburgh); Carelli, M.D.; Bach, C.W.; Killmayer, J.S. Nucl. Sci. Eng.; 64: No. 1, 244-257(Sep 1977).

A study is presented to determine the detailed coolant velocity and temperature profile around the entire rod circumference in liquid-metal fast breeder reactor (LMFBR) core assemblies as well as the detailed radial and circumferential temperature profile in the rod. The digital computer code FATHOM-360 developed to perform the above calculations is described. Fuel, radial blanket, and control assembly rods (both wire-trapped and bare) can be analyzed. Coolant, cladding, and fuel (or absorber) temperature profiles are calculated for uniform and nonuniform heat generation (i.e., accounting for power skew across the pellet) in the rod. Temperature distributions can be calculated for both concentric and eccentric positions of the pellet with respect to the fuel rod cladding. Typical examples of the calculational capabilities of the code are presented. Such capabilities are needed for a reliable design of LMFBR core assemblies and rods to provide detailed cladding temperature profiles and accurately calculate the cladding strain on which the fuel rod lifetime and allowable burnup depend. Overall, a more realistic core thermofluids design is possible by implementing the study presented here.

- 4382 CROSS SECTIONS, ANGULAR DISTRIBUTIONS, AND MAGNETIC SUBSTATE POPULATIONS IN THE $^{23}\text{Na}(n,n'\gamma)$ REACTION. Donati, D.R.; Mathur, S.C.; Sheldon, E.; Barnes, B.K.; Beghian, L.E.; Harihar, P.; Kegel, G.H.R.; Schier, W.A. (Department of Physics and Applied Physics, University of Lowell, Lowell, Massachusetts 01854). Phys. Rev., C; 16: No. 3, 939-957(Sep 1977).

Neutron inelastic scattering cross sections have been determined for six levels in ^{23}Na over the incident neutron energy range 0.52 to 4.23 MeV by measuring the γ -ray production cross sections for 10 deexcitation γ transitions in the $^{23}\text{Na}(n,n'\gamma)$ reaction. γ -ray angular distributions from the first two excited levels were obtained at various incident neutron energies. From these, the magnetic substate populations were evaluated for the 0.440-MeV ($5/2^+$) first excited level and the 2.078-MeV ($7/2^+$) second excited level in ^{23}Na . Comparisons have been made between the distributions and magnetic substate populations calculated from compound-nuclear and direct-interaction theory, the former yielding better agreement in the interpretation of experimental results.

- 4383 TASKS OF AND METHODS FOR IMPURITY CONTROL IN SODIUM LOOPS. Ullmann, H. (Zentralinstitut fuer Kernforschung, Rossendorf bei Dresden (German Democratic Republic)). Kernenergie; 20: No. 9, 269-274(Sep 1977). (In German).

The tasks of impurity control of sodium and inert gas in coolant circuits of sodium-cooled reactors are formulated. From this, the requirements to be met by the measuring methods are derived. The current status of the methods of measurement and analysis are evaluated. Fundamental directions of further development are discussed.

- 4384 DOE TO SPEED REACTOR LICENSING. Power (N.Y.); 121: No. 9, 53-56(Sep 1977).

Deferment of the Clinch River Breeder Reactor development has awakened enthusiasm for alternative nuclear power systems. Most promising are the molten salt reactor developed in Oak Ridge; the HTGR pioneered by General Atomic; and the light-water thorium breeder sponsored by Admiral Hyman Rickover. The Nuclear Regulatory Commission is criticized for its slowness in approving new nuclear plants, thus contributing to 10-year lead times for them. Floating offshore nuclear plants seem to have the best chance of being approved; their lead times are expected to be about six years.

- 4385 INVESTIGATION OF THE PERFORMANCE OF URANIUM-PLUTONIUM MIXED OXIDE FUEL PINS FOR THE REACTOR BN-600. Golovnin, I.S.; Menshikova, T.S.; Bibilashvili, J.K. (Vsesoyuznyi Nauchno-Issledovatel'skiy Inst. Neorganicheskikh Materialov, Moscow (USSR)); Davydov, E.F.; Grjazev, V.M.; Sjuzev, V.N. (Nauchno-Issledovatel'skiy Inst. Atomnykh Reaktorov, Dimitrograd (USSR)). Atomwirtsch., Atomtech.; 22: No. 9, 469-470 (Sep 1977).

A subassembly of experimental fuel pins of the type to be used in BN-600 and using uranium-plutonium mixed oxide fuel had been irradiated up to burnup of 11% heavy atoms and an integrated neutron dose of 6.6×10^{22} n/cm² ($E > 0.1$ MeV) at a maximum clad temperature of 680°C. The most important results were as follows: 1) The dimensional change of the hexagonal wrapper tube (steel 0Ch18N10T) can be attributed to irradiation induced steel swelling of 7.6 vol.% at the neutron dose given above. 2) On an average the diameter of the fuel pins increased by 2% with a maximum of 3%. This diameter increase is essentially due to steel swelling. The diametral cladding deformation caused by radiation creep did not exceed 0.3%. 3) The maximum swelling of the clad material (solution treated steel 0Ch16N15M3B) was found to be 7.3% at about 500°C and at a neutron dose of 6.6×10^{22} n/cm² ($E > 0.1$ MeV). 4) The corrosive interaction between the mixed oxide fuel, the initial O/M-ratio of which was 1.97 to 1.98, and the steel clad was negligible and had no influence on the performance capability of the fuel clad. 5) Strength and plasticity of the clad were reduced due to the irradiation, especially in the high temperature region, but the clad kept its function. 6) The fraction of released fission gas is independent from the initial fuel density and was here found to be approx. 80%. 7) The fuel pins of the BN-600 type with U-Pu-mixed oxide (0.85 UO₂ - 0.15 PuO₂) proved to perform satisfactorily up to a burnup of 11%.

- 4386 STATUS OF FAST BREEDER DEVELOPMENT. Brandstetter, A. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.); Internationale Natrium-Brutreaktor-Bau G.m.b.H., Bergisch Gladbach (Germany, F.R.)). Atomwirtsch., Atomtech.; 22: No. 9, 477-483 (Sep 1977). (In German).

From European conference on particle physics; Budapest, Hungary (4 - 9 Jul 1977). 6 figs.; 2 tabs..

The development of the fast breeder reactor, which has been severely handicapped by the Carter Administration's recent decision, is a must for most other industrialized countries which want to put their energy supply on a firm basis. This development has been pursued with great perseverance and has consumed large amounts of government funds since the early sixties in the United States, the Soviet Union, France, the United Kingdom, Japan and the Federal Republic of Germany. Three prototype reactors are already in operation, four others

in an advanced stage of construction. These many years of development have given fast breeder reactors their high safety standard. However, the main argument in favor of continuing the development of LMFBR's is the need for their commercial introduction in the early nineties. This deadline will perhaps not apply in the United States, which has very large uranium reserves of its own. For the Federal Republic and the developed countries with similar problems in ensuring the continuity of their power supplies, the system encompassing light water reactors as the first generation and sodium cooled fast breeders as the second generation represents a complete and logical energy supply concept. It is the only concept which, based on the present state of the art, permits a specific technology to be adopted for a specific strategy to guarantee the long term continuity of energy supply.

- 4387 FAST BREEDER LICENSING: IS IT UNCONSTITUTIONAL. FULL TEXT OF THE PRESS STATEMENT BY THE UPPER ADMINISTRATIVE COURT AT MÜNSTER. Atomwirtsch., Atomtech.; 22: No. 9, 454 (Sep 1977). (In German).

The statement on the decision to wait for a decision of the Federal Constitutional Court on whether paragraph 7 of the Atomic Energy Act is compatible with the Basic Law is discussed.

- 4388 EXPERIMENTAL INVESTIGATION OF THE FORMATION OF CRITICAL BOILING NUCLEI IN SUPERHEATED SODIUM. Riethmüller, R. (Karlsruhe Univ. (TH) (Germany, F.R.)). Inst. fuer Reaktortechnik. Nucl. Eng. Des.; 43: No. 2, 295-309 (Sep 1977).

This work deals with the problem of boiling nuclei growth up to the critical size in superheated sodium. The sodium was superheated to a certain degree without initiating the onset of boiling. The lifetime of the superheated state - defined as the waiting time - was measured. The experiments were performed with stagnant sodium in a gas-free apparatus. The sodium temperature and superheat were up to 960°C and 360°C, respectively. Waiting times of up to some hours were measured with large scattering of values. Cold trap purifications influenced the waiting time irreversibly. Natural boiling nuclei were unstable. An artificial boiling nucleus (small cylindrical cavity) was inactive. The existence of many cavities in the test chamber surface was demonstrated by a scanning electron microscope. Experimental results are compared with theoretical predictions from known nucleation models. The dynamic model of the formation of collective heterogeneous boiling nuclei agreed qualitatively.

- 4389 DAMPING OF THE TRANSVERSE IMPULSIVE MOTION OF A TIMOSHENKO BEAM RESTING ON A THIN FLUID LAYER. Zehlein, H. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). Nucl. Eng. Des.; 43: No. 2, 373-379 (Sep 1977).

A discrete model for the one-dimensional incompressible transient axial squeeze potential flow within the thin fluid layer between two adjacent LMFBR sub-assemblies caused by the impulsive transverse bending of these beam-type core components is described. The numerical schemes used to determine the time- and space-dependent velocity and pressure fields are outlined. Emphasis is placed on the combined Lagrangian particle/Euler cell approach newly proposed here. In particular two different notions of Lagrangian particles, the convected Lagrangian particle (CLP) and the displaced Lagrangian particle (DLP) are

introduced and evaluated. Some remarks on the coupling technique for the fluid-structure interaction are also given. Preliminary results recommend the DLP because of its higher computational efficiency.

- 4390 TRANSIENT DEFORMATION OF LMFBR CORES DUE TO LOCAL FAILURE: EXPERIMENTAL AND THEORETICAL INVESTIGATION. Beutel, H.; Liebe, R.; Will, H.; Zehlein, H. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)); Rees, N.J.M.; Warren, G.R. (UKAEA Weapons Group, Foulness Atomic Weapons Research Establishment). Nucl. Eng. Des.; 43: No. 2, 381-410 (Sep 1977).

This paper describes an effort to predict the mechanical core deformation caused by local failure within an LMFBR core. These activities are intended to cover all the potential core damage possibilities currently under discussion and analysis. In particular it is shown that the reactor can be scrammed in time under pessimistic-realistic pressure transients and that the damage does not exceed tolerable limits. A special gas generator technique to simulate a fuel coolant explosion has been developed at AWRE Foulness. This has been used to perform the explosion tests needed to demonstrate the safety of the SNR 300 core. A molten fuel-coolant interaction (MFCI) experimental facility, and a drop tower to carry out sub-assembly crushing tests are described. Theoretical studies are presented which use mass-spring-dashpot, lumped parameter-hinge or micro-rigid-lumped-mass models. They simulate the crushing and bending of a single sub-assembly interacting with the coolant as well as the behaviour of a multirow 'spoke' model. For the core analysis only preliminary computational results are presently available which can be compared with the full scale tests in which the fluid pressure did not exceed a 'threshold' of about 100 bar. Parameter studies show the influence of pulse shape, material properties as well as the time integrator. Some of the unanswered questions concern the hydrodynamic feedback of the deformations on the pressure distribution in space and time. Also the behaviour of the highly irradiation-embrittled cores is poorly understood today. Finally, an enhanced energy release package to describe the MFCI must still be added to the reactivity calculation module of a future fast reactor dynamics code.

- 4391 FLOW RUNDOWN EXPERIMENTS IN A SEVEN PIN BUNDLE. Brook, A.J. (UKAEA Reactor Group, Risley); Kaiser, A.; Peppier, W. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). Nucl. Eng. Des.; 43: No. 2, 273-283 (Sep 1977).

A series of experiments has been carried out using an electrically heated seven pin bundle to simulate the conditions under which boiling of the sodium coolant could occur in the event of the loss of power to the circulator pump of a fast reactor, coincident with a failure of the reactor to trip. Although it was not possible to represent the condition of the reactor exactly, nor to continue the tests far into dryout, the results nevertheless give valuable qualitative information on the course of boiling development as well as useful quantitative information against which the predictions of computer codes can be checked. In particular, data have been obtained relating to the incidence of superheat, the location and time to dryout of the residual liquid films, the void fraction within the boiling region, and the types of flow regimes which may be expected within different parts of the boiling region at various stages of the transient.

- 4392 SODIUM BOILING EXPERIMENTS IN A SEVEN-PIN BUNDLE: FLOW PATTERNS AND TWO-PHASE PRESSURE DROP. Kaiser, A.; Peppier, W. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). Nucl. Eng. Des.; 43: No. 2, 285-293 (Sep 1977).

Two series of quasi-steady state sodium boiling experiments have been carried out in an electrically heated seven-pin bundle. The power levels (130-170 and 30-40 W/cm²) and other test conditions were selected to correspond to the core and radial breeder zones of a typical LMFBR. The test procedures involved the gradual reduction of mass flow rate through the bundle in a simulation of the consequences of a slowly growing blockage in the lower part of a reactor subassembly. By this means it was possible to study the development of quasi-steady state boiling up to the onset of permanent dryout. The results obtained provide information of flow regimes in the two-phase region, vapour qualities and flow rates at which cooling of the bundle can be effectively maintained, and the ultimate incidence of dryout. A relation for the two-phase pressure drop multiplier obtained from adiabatic pressure drop measurements in this geometry is given and compared with earlier results from single-channel geometry tests.

- 4393 SODIUM BOILING EXPERIMENTS IN A 7-PIN BUNDLE: INVESTIGATIONS INTO SOME SAFETY ASPECTS OF THE KNK II DRIVER ZONE. Kaiser, A.; Peppier, W.; Straka, M. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). Nucl. Eng. Des.; 43: No. 2, 411-417 (Sep 1977).

A series of sodium boiling experiments, in which the thermohydraulic characteristics of KNK II driver subassemblies were simulated, has been carried out for the purpose of studying the effects of two types of cooling disturbances: rapid flow interruption and those flow reductions which develop rather gradually. Information about the spatial and temporal development of boiling, voidage and dryout was obtained. Furthermore the feasibility of individual subassembly temperature monitoring has been investigated as well as that of two integral boiling detection methods based on acoustic noise and reactivity measurements.

- 4394 ANALYSIS OF HYPOTHETICAL OVERPOWER ACCIDENTS IN THE SNR-300 MARK 1A CORE WITH MODELING OF FAILURE INCOHERENCE. Royl, P.; Kuczera, B. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). Nucl. Eng. Des.; 43: No. 2, 239-248 (Sep 1977).

Incoherency effects due to the statistical nature of fuel rod failure have been assessed for transient overpower (TOP) accidents with different initiating ramp rates. Characteristic single subassembly feedbacks from sodium voiding, fuel injection, and fuel sweep-out after failure were calculated and superimposed incoherently. Effective feedback functions were thus generated for characteristic subassembly groups (channels) and used as tabular reactivity input to corresponding multi-channel whole core TOP simulations using the HOPE computer model. Analyses were performed for the SNR-300 Mark 1A end of life (EOL) core with initiating ramp rates of 0.15, 0.5 and 5 S/sec. The results are compared with those from a calculation with coherent subassembly behavior in each channel. It is concluded for the mild transient that irrespective of the assumed axial failure location, negative fuel sweep-out reactivity from early failed subassemblies will prevent the accident from reaching a prompt

critical stage and lead to a gradual shutdown if the assessed failure incoherencies are applied. For the higher ramp rates (>0.5 S/sec) the assessed incoherencies will not always lead to an early shutdown, but they can considerably reduce the potential to accumulate high ramp rates in the initiating phase of TOP accidents.

- 4395 IN-PILE SODIUM LOOP FOR INVESTIGATION OF LOCAL COOLING DISTURBANCES IN LMFBR'S. Kramer, W.; Schlesiak, K.; Schmidt, L. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)); Vanmassenhove, G.; Verwimp, A. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)). Nucl. Eng. Des.; 43: No. 2, 343-351 (Sep 1977).

In the safety analysis of Liquid Metal Fast Breeder Reactors, investigations of the fuel element behavior under local off-normal cooling conditions and the possible failure propagation are of special interest. In a common program, called 'Mol 7C' the Gesellschaft fuer Kernforschung, Karlsruhe, and the Centre de l'Etude de l'Energie Nucleaire/Studiecentrum voor Kernenergie, Mol, are performing related in-pile experiments in a sodium loop in the BR 2-reactor. The test section contains a 37-rod bundle of fresh UO_2 -fuel. A local blockage within the fuel bundle will initiate a certain local damage to a few rods. The experiments are expected to obtain important informations with respect to the problems of pin to pin propagation and the long term behaviour of a fuel bundle with defect pins. The in-pile part of the loop contains the fully integrated primary sodium circuit. Total heat removal capacity is about 700 kW. The equipment for the first experiment is nearly manufactured. The first experiment will start in the beginning of 1977. At first three experiments are planned.

- 4396 THEORETICAL INTERPRETATION OF SCARABEE SINGLE PIN IN-PILE BOILING EXPERIMENTS. Struwe, D.; Bottoni, M.; Fries, W.; Elbel, H.; Angerer, G. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). Nucl. Eng. Des.; 43: No. 2, 259-271 (Sep 1977).

In the framework of LMFBR safety analysis a theoretical interpretation of some of the most representative of the single pin experiments of the in-pile SCARABEE project has been performed from both viewpoints of thermohydraulic and fuel behaviour using the computer codes CAPRI-2 and SATURN-1. The analysis is aimed at investigating the pin behavior from the preirradiation history, through the observed sequence of events following a coolant mass flow reduction from boiling inception up to pin breakdown. A comparison of theoretical results with experimentally recorded data has allowed a deeper insight into the peculiar features of the experiments and enabled a valuable code verification.

- 4397 STEELS FOR THE PRIMARY CIRCUITS OF THE SODIUM COOLED BREEDER REACTOR. Weisz, M.; Berge, P.; Rabbe, P. (Cent d'Etud Nucl, Sclay, Fr.). Mater. Tech.; No. 9-10, 477-491 (Sep-Oct 1977). (In French and English).

Certain factors that must be taken into account in determining the allowable stresses in austenitic stainless steels for use in the primary circuits of sodium-cooled breeder reactors are reviewed. These include the effects of heat-to-heat variations on stress values determined from tensile and creep tests, the effect of the sodium medium on the creep and fatigue properties of the steels, including the intrinsic effects due to the presence of a very pure liquid metal and the effects due to

the transfer of alloying elements, particularly carbon. Also discussed are the effects of long holdings at high temperatures on the impact strength and intergranular corrosion sensitivity of the steels. Finally, the risks of cracking in the heat affected zones during welding and in service are assessed. 18 refs.

- 4398 SELECTION OF STEAM GENERATOR MATERIALS FOR SODIUM COOLED BREEDER REACTORS. Berge, P. (Electr de Fr). Mater. Tech.; No. 9-10, 493-510 (Sep-Oct 1977). (In French and English).

This paper deals with the determination of the permissible stresses for steels considered for fast breeder steam generator tubes. The mechanical properties of the tubes at elevated temperatures are discussed, as well as research on their fatigue resistance, the influence of sodium on their mechanical properties, and the effect of heat treatment on their sensitivity to intergranular corrosion. Also discussed are the weldability of the steels employed and the weld properties, the decarburization of ferritic steels, and corrosion in the water-steam circuit under normal operating conditions and during accidents. 45 refs.

- 4399 NOTCH DUCTILITY OF ALLOY 800 FOR THE STEAM GENERATORS OF BREEDER REACTORS. Cozar, R.; Rouby, M.; Lagadec, H.; Julien, M. (Creusot-Loire, Fr.). Mater. Tech.; No. 9-10, 511-524 (Sep-Oct 1977). (In French and English).

The aim of this study was to define the notch ductility at elevated temperature of Alloy 800 as specified for the Superphenix steam generator tube. The materials investigated, which are representative of the tube, were processed in various ways. It was found that homogeneous precipitation of the γ' phase improves their creep strength but reduces their ductility. The formation of this phase is governed by chemical composition, and in particular by the Ti+Al sum content. The limiting value of 0.65 to 0.70% for this content seems to be the threshold for noticeable effects of the γ' phase. The results of this study indicate that materials that show no trace of this homogeneous precipitation of the γ' phase are immune to the notch effect. Alloys having a Ti+Al sum content of between 0.40 and 0.65% are in this category. The ductility of these materials remains satisfactory even in the 10% and 15% cold worked state, after tests at 550°C. In comparison, a material with a Ti+Al sum content of 0.94% is very susceptible to the notch effect at 550°C; notched creep specimens fractured after 650 hr, whereas unnotched specimens resist periods of 2500 hr under a stress of 250 N/mm². 10 refs.

- 4400 UTILIZATION PROPERTIES OF SOME SPECIAL STEELS FOR SODIUM COOLED BREEDER REACTORS. Mathern, G.; Crolet, J.L.; Seraphin, L.; Tricot, R. (Pechiney Ugine Kuhlmann, Fr.). Mater. Tech.; No. 9-10, 525-542 (Sep-Oct 1977). (In French and English).

It is shown that 2.25% Cr-1% Mo low-alloy steels stabilized by niobium or by niobium/vanadium offer good resistance to corrosion by water and liquid sodium. Their mechanical properties can be adjusted by varying the heat treatment. The creep properties of niobium/vanadium stabilized steel are higher than those of steel stabilized with niobium alone if the quenching treatment is sufficiently severe to obtain an entirely bainitic structure. The behavior of niobium-vanadium stabilized ferritic steels with 9-12% chromium content is similar to that of steels of the previous

category. They withstand oxidation by steam up to higher temperatures; their hardenability is greater than that of low alloy steels. 18-10 or 18-12 Mo austenitic steels offer poor resistance to stress-corrosion in water and can, therefore, only be used in dry steam. However, their resistance to liquid sodium is good. Their creep properties can be optimized by adding boron and by controlling the Ti/C stabilizing ratio for steels containing titanium. They are very easy to weld providing certain precautions are taken to prevent hot cracking. The stress-corrosion resistance of Incoloy 800 is quite superior to that of the previous steels, which means that it can be used in water or wet steam. The alloy can be rendered immune to intergranular corrosion by suitable heat treatment. 20 refs.

- 4401 PRODUCTION AND CONTROL OF STAINLESS STEEL CLADDING TUBES FOR BREEDER REACTORS. Jackson, M.; Spriet, B. (Soc. Vallourec, Fr). Mater. Tech.; No. 9-10, 595-603 (Sep-Oct 1977). (In French and English).

A description is given of the manufacture and inspection of tubes for the cladding of fuel elements used in fast reactors. Particular attention is paid to nondestructive testing of the cladding tubes by eddy-current and ultrasonic inspection, and also by metrological inspection of the outside diameter and wall thickness. 3 refs.

- 4402 EFFECTIVE METHOD FOR ESTIMATING MULTIPLE CONTROL ROD WORTHS IN LARGE FAST REACTORS. Mitani, H. (Jpn At Energy Res Inst, Tokai-mura, Ibaraki-ken). J. Nucl. Sci. Technol. (Tokyo); 14: No. 9, 640-651 (Sep 1977).

A method for estimating the multiple control rod worth in large fast reactors has been developed using the higher order perturbation technique. With this method, physical significance of the interaction effect between multiple control rods can be understood by decomposing it into independent components with different physical processes. Furthermore, a new technique is proposed, whereby the interaction effect between multiple control rods can be estimated from a combination of the interaction effects between two or three control rods depending on the strength of the effect. The results show that the interaction effect between multiple control rods in prototype fast reactors can be estimated from a combination of the effects between 2 control rods with a good accuracy. The numerical examinations for a 1,000 MWe fast reactor showed that the interaction effect is too large to apply the present method for large fast reactors. To overcome this difficulty, a new concept of 'quasi-control-rod' is introduced in the method. With this improved method, the worths of 19 and 18 control rods in the 1,000 MWe fast reactor are estimated with accuracies of $\pm 2\%$. 18 refs.

- 4403 DISSOLUTION RATIO OF HYDROGEN GENERATED BY SMALL-LEAK SODIUM-WATER REACTION. Nei, H. (Power React and Nucl Fuel Dev Corp, Oarai Eng Cent, Ibaraki-ken, Jpn). J. Nucl. Sci. Technol. (Tokyo); 14: No. 9, 652-660 (Sep 1977).

A study, both theoretical and experimental, was undertaken on the hydrogen dissolution ratio in liquid sodium, with the view of estimating the quantity of hydrogen that dissolves into sodium before reaching the free surface, following the occurrence of sodium-water reaction in a steam generator. Superheated steam was injected into the sodium in a reaction vessel of configuration

simulating the steam generator. The hydrogen dissolution ratio was determined from analysis of the responses obtained on a hydrogen detector installed downstream of the reaction vessel. A theoretical model was assumed for estimating the hydrogen dissolution ratio, which yield results in good agreement with the experimental data, in respect of the rising tendency of the dissolution ratio with decreasing leak rate and with rising sodium temperature. 14 refs.

- 4404 SPLITTING METHOD FOR COMPUTING COUPLED HYDRODYNAMIC AND STRUCTURAL RESPONSE. Ash, J.E. (Argonne National Lab., IL). Appl. Math. Model.; 1: No. 6, 333-338 (Sep 1977).

A numerical method is developed for application to unsteady fluid dynamics problems, in particular to the mechanics following a sudden release of high energy. Solution of the initial compressible flow phase provides input to a power-series method for the incompressible fluid motions. The system is split into spatial and time domains which lead to the convergent computation of a sequence of elliptic equations. Two sample problems are solved, the first involves an underwater explosion and the second the response of a nuclear reactor containment shell structure to a hypothetical core accident. The solutions are correlated with experimental data.

- 4405 SUBASSEMBLY EXPERIMENTS AND A COMPUTER CODE TO ANALYZE THE DYNAMIC CORE DEFORMATION DURING LOCAL FAILURE PROPAGATION. Liebe, R. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). Nucl. Eng. Des.; 43: No. 2, 353-371 (Sep 1977).

Safety investigations for LMFBRs have to consider local failure situations in one fuel element which may escalate to a hypothetical CDA. Such initiating events could produce high pressure pulses in a single subassembly which may expand and rupture the wrapper as well as load adjacent elements impulsively. The associated nonlinear dynamic core deformation problem is treated in this paper. In particular the multirow structural dynamics code COERE-1 and underlying mechanical models are described. Each subassembly is simulated by an equivalent system of point masses and nonlinear coupling springs. The motion of the coolant layer between the elements is treated by an incompressible, non-stationary frictional flow model. In order to obtain realistic code input four types of static single subassembly deformation experiments are described which provided strongly nonlinear load deformation characteristics.

- 4406 MODEL FOR FORMATION AND GROWTH OF LOCAL BLOCKAGES IN GRID SPACED FAST SODIUM COOLED BREEDER FUEL ELEMENTS. Schultheiss, G.F. (Karlsruhe Univ. (TH) (Germany, F.R.)). Inst. fuer Reaktortechnik. Nucl. Eng. Des.; 43: No. 2, 329-341 (Sep 1977).

The studies of the formation of local cooling flow disturbances in fast sodium breeder reactors are intended to furnish information, which can be used for safety assessments, about the attainable extent of blockages and the time required for their growth. The formation of major local blockages in normal fuel element geometries will mainly be caused by fuel expelled after a cladding tube failure. A model of the screening process is developed considering the results of in-pile experiments, thermohydraulic studies and simulation experiments of the growth of blockages. The model is limited to the initial phase in which the largest free cross sections

in a subchannel with a spacer grid geometry must be plugged up to constitute a so-called 'fundamental blockage'. Parameter studies on the model show that events constituting the formation of a blockage occur slowly.

- 4407 EFFECT OF THE BR-5 REACTOR OPERATING CONDITIONS ON THE DEVELOPMENT OF VACANCY POROSITY IN THE 1KH18N9T STEEL. Porollo, S.I.; Shcherbak, V.I.; Aristarkhov, N.N.; Bykov, V.N.; Dmitriev, V.D.; Mamaev, L.I. At. Energ. (USSR); 43: No. 3, 206-207 (Sep 1977). (In Russian).

FAST NEUTRONS; FUEL CHANNELS; FUEL ELEMENTS; HIGH TEMPERATURE; MICROSTRUCTURE; NEUTRON FLUENCE; PLUTONIUM DIOXIDE; POROSITY; REACTOR OPERATION; SBR-5 REACTOR; SWELLING; TEMPERATURE DEPENDENCE; URANIUM CARBIDES; VACANCIES; CHROMIUM-NICKEL STEELS; PHYSICAL RADIATION EFFECTS

- 4408 DEVELOPMENT OF A NEW TUBE-TO-TUBESHEET WELDING TYPE FOR FBR'S HEAT EXCHANGERS. Mase, N.; Kawara, M.; Shibato, E.; Kashiwakura, K. (Hitachi Ltd., Ibaraki (Japan). Hitachi Works). Hitachi Hyoron; 59: No. 9, 747-752 (Sep 1977). (In Japanese).

The intermediate heat exchanger for exchanging heat between primary and secondary sodium and the steam generator are required to guarantee the performance and the reliability of construction over long term under the environment of high temperature sodium peculiar to a FBR. With the increase of power output of FBR plants, the number of heating tubes and the size of tube plates of the intermediate heat exchangers of shell and tube type increase. In order to improve the reliability and the production process of the tube to tube plate welding, a new method was developed, according to which heating tubes are inserted into tube plate holes by about their thickness, and the welding of perfect penetration is carried out internally with an automatic TIG welder. At the same time, in order to confirm the reliability of the welded joints by this method, the tests for evaluating the strength for short time and long term were carried out. It was confirmed that the satisfactory performance of the welded joints was able to be obtained, and the method would be applicable to actual heat exchangers. As for the nondestructive inspection of welded joints, the radiographic method was established, and ultrasonic and eddy current flaw detection methods are being developed now.

- 4409 DISSOLUTION RATIO OF HYDROGEN GENERATED BY SMALL-LEAK SODIUM-WATER REACTION. Nei, H. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). J. Nucl. Sci. Technol. (Tokyo); 14: No. 9, 652-660 (Sep 1977).

A study, both theoretical and experimental, was undertaken on the hydrogen dissolution ratio in liquid sodium, with the view of estimating the quantity of hydrogen that dissolves into sodium before reaching the free surface, following the occurrence of sodium-water reaction in a steam generator. Superheated steam was injected into the sodium in a reaction vessel of configuration simulating the steam generator. The hydrogen dissolution ratio was determined from analysis of the responses obtained on a hydrogen detector installed downstream of the reaction vessel. The hydrogen dissolution ratio was found to vary from 5 to 15% over a leak rate ranging of 3 -- 10 g/sec, as determined both from the transient response data given by the hydrogen detector and from the corresponding data obtained from tests on changes in hydrogen concentration upon complete mixture with the

entire sodium inventory. A theoretical model was assumed for estimating the hydrogen dissolution ratio, which yield results in good agreement with the experimental data, in respect of the rising tendency of the dissolution ratio with decreasing leak rate and with rising sodium temperature.

- 4410 NOTCH DUCTILITY OF ALLOY 800 FOR LIQUID METAL FAST BREEDER REACTOR STEAM GENERATOR. Cozar, R.; Rouby, M. (Societe Creusot-Loire, 75 - Paris (France)); Lagade, H.; Julien, M. (Fives-Cail-Babcock, 75 - Paris (France)). Mater. Tech.; No. 9, 511-524 (Sep-Oct 1977). (In French and English).

From 2. Colloquium on special steels and nuclear energy; Paris, France (14 - 15 Dec 1976).

The grade selected for the manufacture of heat-exchange tubes for fast breeder reactor steam generators is an Ni Fe Cr alloy (2 4 NC 33.21). This choice is vindicated by the operating characteristics of this material. Are in particular studied: the alloy mechanical properties (low temperature creep - 500 to 600°C), the internal stress relaxation, the determination of the alloy ductility at low and medium temperatures, the precipitation of intermetallic phases, the structure to product thermomechanical history relationship.

- 4411 PRODUCTION AND CONTROL OF STAINLESS STEEL CLADDING TUBES FOR BREEDER REACTORS. Jacson, M. (Vallourec, 75 - Paris (France)); Spriet, B. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. de Developpement des Elements Combustibles). Mater. Tech.; No. 9, 595-603 (Sep-Oct 1977). (In French, English).

From 2. Colloquium on special steels and nuclear energy; Paris, France (14 - 15 Dec 1976).

The reliability of a cladding tube in a LMFBR depends, for the most part, on its capability of keeping its original shape, even under extremely heavy duty conditions. A tremendous effort has been conducted in two different ways, to attain this purpose: definition of the metallurgical characteristics of the cladding tube material according to its mechanical properties in high temperature conditions and its properties under neutronic action; an improved quality of the tubes by methodical and controlled processing conducted at Vallourec's plant in Montbard in a specially developed workshop for this type of tubes. The raw material controls, the tubes production scheduling in hot and cold conditions, together with the non-destructive and destructive tests are defined both by Vallourec and the C.E.A.

- 4412 UTILIZATION PROPERTIES OF SOME SPECIAL STEELS FOR SODIUM COOLED BREEDER REACTORS. Mathern, G.; Crolet, J.L.; Seraphin, L.; Tricot, R. (Ugine Aciers, 73 (France)). Mater. Tech.; No. 9, 525-542 (Sep-Oct 1977). (In French and English).

From 2. Colloquium on special steels and nuclear energy; Paris, France (14 - 15 Dec 1976).

Among the liquid metals, sodium constitutes a particularly effective coolant for removing large amounts of heat produced in the core of fast neutron breeder reactors. The following is examined: the mechanisms of corrosion and decarburization by liquid sodium in movement; the interest of stabilized ferritic steels used in mixed circuits or in homogeneous circuits; the respective positions of austenitic stainless steels from the viewpoint of utilization and working properties, such as

corrosion resistance and creep strength.

- 4413 STEELS FOR THE PRIMARY CIRCUITS OF THE SODIUM COOLED BREEDER REACTOR. Weisz, M. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Service de Recherches Metallurgiques Appliquees); Berge, P. (Electricite de France, 77 - Ecuellles. Dept. Etudes des Materiaux); Rabbe, P. (Societe Creusot-Loire, 42 - Unieux (France)). Mater. Tech.; No. 9, 477-491 (Sep-Oct 1977). (In French and English).

From 2. Colloquium on special steels and nuclear energy; Paris, France (14 - 15 Dec 1976).

This paper describes the effort made in the working group E.D.F.-C.E.A.-Industry, on the molybdenum austenitic steels in order to evaluate the best compromise between the various properties, considering the temperature conditions (350 - 550°C) and the mechanical working conditions: creep resistance, especially in the low strain range (0,1%); modification of toughness due to long holding time at high temperature; satisfactory resistance to intergranular corrosion; weldability; properties of the metal deposited in the heat affected zones; absence of cracking at high temperature.

- 4414 EXPERIMENTS WITH SLIDING SUSPENSION IN LIQUID SODIUM. Matul, O. (Vyzkumny Ustav Zavodu Energetického Strojirenstvi, Brno-Slatina (Czechoslovakia)). Strojirenstvi; 27: No. 9, 565-569 (Sep 1977). (In Czech).

Selected sliding pairs were tested as a model of sliding assembly on a model equipment in liquid sodium. The sliding pair consisted of a pellet 20 mm in diameter and a thickness of 8 mm with a ring 60 mm in diameter and a hole 45 mm in diameter and a width of 25 mm which was mounted on a rotating shaft. The peripheral velocity of the rings was 1.25 m/s. The pellets were pressed to the rings by an intermittent force of 1 to 1.7 kp at a frequency of intermittence 4 Hz. The sliding pairs were manufactured by the flame deposition of materials of the type Cr-Ni-B-Si, Cr-Ni-B-Si-WC, WC-Ni, WC-Co, Ti-steel on a steel semi-finished product (Czechoslovak Standard 17 246). The surface of the sliding pairs was studied using x-ray spectral microanalysis, electron microscopy and metallography. The short-term tests of sliding pairs in liquid sodium showed small wear, a good compactness of the deposited layer with the base material before exposure and after exposure in liquid sodium and resistance to corrosion by sodium. Only materials of the type WC-Co and WC-Ni did not show the expected properties.

- 4415 CONSTRUCTION OF SOVIET FAST REACTOR BN-600. Suransky, F. Jad. Energ.; 23: No. 9, 352-357 (Sep 1977). (In Czech).

A sectional view is shown of the integral configuration of the 3rd unit reactor in the Beloyarsk nuclear power plant. The reactor vessel is a cylinder 12.8 m in diameter and 12.6 m in height. In view of overpressure in the vessel (40 kPa) the wall thickness is 30 to 40 mm. The reactor core contains 370 hexagonal fuel elements. Each element consists of 127 pins of an outer diameter of 6.9 mm. 27 positions are taken by regulating and scram rods. The fuel reserve in the core and the efficiency of reactivity control permits reactor operation for about 150 days such that one third of the fuel elements is exchanged during refuelling. A block diagram is shown of the power plant heat generating system. Core cooling is ensured by three circuits, i.e., the

sodium primary and secondary circuits and one water and steam circuit. The progress of the power plant construction is briefly indicated.

- 4416 CORROSION OF REFRACTORY BORIDES IN FUSED SODIUM. Arabei, B.G.; Glukhov, V.A.; Markov, Y. (All-Union Sci-Res Inst of Aviat Mater, USSR). Prot. Met. (Engl. Transl.); 13: No. 5, 492-494 (Sep-Oct 1977).

Corrosion of chromium, tantalum, and hafnium diborides, samarium, gadolinium, and europium hexaborides, and composites of europium hexaboride with diborides of these metals was tested. As in the case of the diboride structure, no appreciable changes were detected in the structure of the $\text{MeB}_2\text{-EuB}_6$ composites. These composite materials can be recommended for use with sodium up to 800°C for periods of up to 5000 h.

- 4417 CORROSION RESISTANCE OF ALLOYS OF THE NIOBIUM-ZIRCONIUM-OXYGEN SYSTEM IN MOLTEN LITHIUM. Arakelov, A.G.; Vavilova, V.V.; Gekov, A.G.; Zel'tser, A.M. (Acad of Sci of the USSR, A.A. Baikov Inst of Metall, Moscow). Sov. Mater. Sci. (Engl. Transl.); 13: No. 5, 503-506 (Sep-Oct 1977).

Investigations were directed toward studying the effect of the phase composition, ratio of zirconium and oxygen in niobium-base alloys, and the heat treatment of alloys of the Nb-Zr-O system on corrosion resistance in molten lithium. The investigation of the interaction of alloys of the Nb-Zr-O system with lithium was conducted at 400°C by the method of full immersion of the specimen in a niobium ampul with lithium previously heated to this temperature. The method of microstructural analysis was used in studying the alloys, along with measurements of microhardness and resistivity, x-ray crystallographic analysis, and ion analysis. A comparison of the results of studies of the behavior of alloys of the binary system niobium-oxygen and the ternary system of niobium-zirconium-oxygen shows that the character of lithium damage to alloys with oxygen in the ternary system is less sharp, although, as with the binary system there is a reduction in microhardness and an increase in resistivity for alloys held in lithium. It must be noted that even in the section Nb-ZrO₂ corrosion takes place after annealing at 1500°C. However, the corrosion is considerably less despite the fact that a ratio Zr:O=1:2 is observed in the alloys. 14 refs.

- 4418 SELECTION OF STEAM GENERATOR MATERIALS FOR SODIUM COOLED FAST BREEDERS. Berge, P. (Electricite de France, 77 - Ecuellles. Dept. Etudes des Materiaux). Mater. Tech.; No. Special No. 9, 493-510 (Sep-Oct 1977). (In French and English).

From 2. Colloquium on special steels and nuclear energy; Paris, France (14 - 15 Dec 1976).

The sodium water heat exchangers are now considered as the stumbling block in the development of liquid metal cooled fast breeders, due to the risk of sodium-water reactions. The selection of the materials for these tube-bundles has been very broad, for the different existing, or in-project, reactors in the world: low alloy 2 1/4 Cr - 1 Mo steels (unstabilized or stabilized); 9 Cr - 1 Mo ferritic steel; 18 Cr - 10 Ni austenitic stainless steels; alloy 800. One can also add other ferritic steels, as 9 Cr - 2 Mo stabilized, which are studied for this application. In the framework of the E.D.F.-C.E.A. working group a major effort was undertaken to study the characteristics of

these various materials with respect to the main criteria governing construction of the tube bundles and their performance in service: mechanical characteristics at high temperature; fabrication and welding; behavior with respect to mass transfer in sodium; carburization and decarburization; corrosion resistance. The main lines and results of this program are described.

- 4419 CONTROL ROD AND/OR INSTRUMENT TREE ASSEMBLY. Noyes, R.C.; Dupen, C.F.G.; Barrus, D.M.; Siegel, E.A. (to Combustion Engineering, Inc.). US Patent 4,045,283. 30 Aug 1977. Filed date 30 Dec 1974. 14p.

Disclosed is an improved core servicing apparatus for a nuclear reactor having a vessel, a head for the vessel, one or more penetration openings in the head and a core within the vessel of the type having an array of fuel assemblies and including apertures in the array for neutron absorbers. The servicing apparatus includes a support column adapted to be supported by and depend from the head in registry with a penetration opening; a plurality of core servicing elements which may include neutron absorber actuating means and/or instrument packages are supported by the support column; and means are provided for laterally displacing at least some of the servicing elements relative to the support column between a laterally compact and a laterally expanded configuration. In one embodiment, instruments may be located at the lower end of long resilient cables and are longitudinally movable within and beyond guide channels supported by the support column. The guide channels are directed outward at their lower ends to direct a descending instrument and cable outwardly. 9 claims, 18 figures.

- 4420 MAINTENANCE SYSTEM FOR IMMersed SEALS, SPECIFICALLY FOR NUCLEAR REACTORS. Poindexter, A.M.; Ricks, H.E. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). French Patent 2,363,862/A/. 30 Aug 1977. Priority date 31 Aug 1976, United States of America (USA), 22p. (In French).

The invention concerns the immersed seals of nuclear reactors and specifically a maintenance system for the immersed seals of the revolving closing plugs of liquid metal breeder nuclear reactors. A liquid sodium immersed joint may be located at a given place or be surrounded by heating elements so that the sodium stays liquid whilst the reactor is working. In other cases, the sodium in the immersed seal is allowed to solidify whilst the reactor is working, thereby increasing the efficiency of the seal. At all events, the sodium must be in a liquid state during reloading with fuel to enable the plug to turn. The invention consists in fitting an ultrasonic transducer to the closure head of the reactor vessel so that the vibration emitting surface directs these vibrations towards the immersed seals so as to detach the deposits of impurities on them and ensure the wetting of the metal surfaces of which they are formed. Additionally, an envelope that can be placed around the ultrasonic transducer in conjunction with a suction appliance provides a mechanism through which the impurities can be removed from the area of the immersed seal.

- 4421 FUEL ROD FOR LIQUID METAL COOLED NUCLEAR REACTORS. Vinz, P. (to Messerschmitt-Boelkow-Blohm G.m.b.H., Muenchen (Germany, F.R.)). German (FRG) Patent 2,449,597/C/. 25 Aug 1977. 5p. (In German).

This fuel rod should, in spite of increased

thermal surface loading, exclude the formation of steam bubbles and local overheating with certainty, which otherwise occur, on non-wetting cans. The can, made as is usual for sodium cooled reactors of precious metal with nitrated non-wetting surface layer is wound with a compatible precious metal wire of 0.5 to 0.05 mm diameter. The prestressed winding leaves a space of about 1/10 diameter between turns. The turns are fixed to the can by diffusion welding giving intensive thermal contact. The windings produce the decisive effects: as the precious metal wire is completely wetted by hot sodium, the danger of forming 'hot spots' at the can surface which is associated with a non-wetted can due to defective heat transfer is avoided. Also, the back channel formed between 2 windings, i.e. around the whole winding forms a nucleus for bubbles in the space between the wetted wire surface and the non-wetted can surface.

- 4422 LIQUID METAL COOLED FAST BREEDER NUCLEAR REACTORS. Durston, J.G. (to United Kingdom Atomic Energy Authority). US Patent 4,043,866. 23 Aug 1977. Priority date 6 Dec 1973, United Kingdom of Great Britain and Northern Ireland (UK), 4p.

A liquid metal fast breeder nuclear reactor of the pool kind is described. The intermediate heat exchanger depends from the roof of the primary vessel in a housing and is immersed in the reactor coolant. A gas pocket seals the passage between the heat exchanger shell and the housing to prevent liquid metal by-passing the tube bundle.

- 4423 STRUCTURE FOR REDUCING CONVECTION CURRENTS WITHIN THE PRESSURE VESSEL OF A FAST REACTOR. Benque, J.P.; Lallement, J.; Lambert, P. (to Commissariat a l'Energie Atomique). US Patent 4,043,867. 23 Aug 1977. Priority date 3 Sep 1974, France, 4p.

A hollow metallic ring structure is secured to the periphery of the diagrid support and located within an internal annular space formed between the lateral neutron shield of the reactor core and an inner tank placed within a primary vessel. The ring structure is smaller in width than the annular space, has a maximum height equal to that of the reactor core, and is provided with partition-walls and holes for producing pressure drops and reducing convection currents.

- 4424 HEAT UTILIZATION OF FUTURE REACTOR SYSTEMS. Heller, A.; Minder, R. (Electro-Watt Engineering Services Ltd., Zurich (Switzerland)). Bull. Schweiz. Elektrotech. Ver.; 68: No. 16, 839-843 (13 Aug 1977). (In German).

From SEV/VSE meeting on heat utilization of nuclear power stations; Zurich, Switzerland (28 Jun 1977).

The types of nuclear reactor systems suitable for heat utilization are reviewed comparing the existing designs such as the PWR (pressurized water reactor) and BWR (boiling water reactor) with possible future reactor systems including the LMFBR (liquid metal fast breeder reactor) and HHT (high temperature helium turbine reactor). Next to the high pressure = high temperature advanced technology systems, the prospects of low technology solutions are discussed such as the Swedish SECURE (safe environmentally clean urban reactor) project designed as a pure heat producing plant.

- 4425 DRIVE FOR CONTROL ROD OF A NUCLEAR

REACTOR. Schabert, H.P.; Pawlitzki, R. (to Siemens A.G., Berlin (Germany, F.R.); Siemens A.G., München (Germany, F.R.)). German (FRG) Patent 1,813,293/C/. 11 Aug 1977. 3p. (In German).

For this drive, a control rod driven axially in the reactor core is uncoupled from its driving rod, and interlocked in the lowest position without involving moving parts. The control rod is coupled to one of the driving rods coming from above by a type of bayonet coupling by a short lifting and turning movement, this moves the control rod up and down in a profiled rod channel, e.g. a hexagonal tube in a sodium-cooled reactor, with hexagonal fuel element. Six cams on the circumference of the control rod guide it in the hexagonal tube, and prevent turning over the whole travel of, for example, 8 metres. Only in the shutdown condition of the reactor can window shaped cutouts in the 6 corners of the guide tube produce tangential rotation of the cam with the control rod. Thus the control rod can be interlocked in this position and can be prevented from lifting due to coolant flow.

- 4426 LIQUID METAL COOLED REACTOR PROVIDED WITH A SYSTEM LIMITING THE CORE DISPLACEMENT WITH RESPECT TO CONTROL RODS. Seed, G.; Hodgson, D.; Grime, C.J. (to Nuclear Power Co. (Whetstone) Ltd. (UK)). French Patent 2,361,721/A/. 11 Aug 1977. Priority date 12 Aug 1976, United Kingdom of Great Britain and Northern Ireland (UK), 5p. (In French).

This invention covers the construction of liquid metal cooled nuclear reactors. Under the invention, the core is immersed in a coolant mass contained in a primary vessel. This vessel is suspended to the roof of a containment pit and is surrounded by a leak proof jacket. The control rods are supported by the roof and can be introduced into the core. The core is supported from the wall of the lower part of the primary vessel, a device being provided in the space between this vessel and the leak proof jacket to restrict the amount of downward bending towards the lower part of the primary vessel. Hence, in the event of the wall of the reactor vessel bending, which would bring about a lowering of the reactor core in relation to the control rods, the leak proof jacket forms a secondary supporting structure for the core and restricts its movement.

- 4427 LIQUID METAL COOLED REACTOR PROVIDED WITH A SYSTEM LIMITING THE CORE DISPLACEMENT WITH RESPECT TO CONTROL RODS. Seed, G.; Hodgson, D.; Grime, C.J. (to Nuclear Power Co. (Whetstone) Ltd. (UK)). French Patent 2,361,722/A/. 11 Aug 1977. Priority date 12 Aug 1976, United Kingdom of Great Britain and Northern Ireland (UK), 9p. (In French).

This invention concerns the construction of liquid metal cooled nuclear reactors. Under the invention the reactor includes a core immersed in a coolant mass contained in a reactor jacket suspended from the roof of a containment pit and control rods supported by the roof can be introduced into the core. This core is maintained as from the wall of the reactor jacket by a platform and a second supporting structure is provided for the core to be used should the platform collapse. In the event of or partial failure of the platform or of its support, the resulting dropping of the reactor core with respect to the control rods is limited in full safety by the second supporting structure.

- 4428 LIQUID METAL COOLED NUCLEAR REACTOR. Seed, G.; Hodgson, D.; Grime, C.J. (to

Nuclear Power Co. (Whetstone) Ltd. (UK)). French Patent 2,361,720/A/. 11 Aug 1977. Priority date 12 Aug 1976, United Kingdom of Great Britain and Northern Ireland (UK), 9p. (In French).

Under this invention, in a nuclear reactor cooled by a liquid metal comprising a primary vessel suspended to the roof of a concrete containment pit, said reactor vessel containing the coolant mass includes an upper open end and is suspended from the roof by ties, this reactor vessel and ties being enveloped in a leak proof jacket sealed to the roof of the pit along its edge and in which there is an inert gas to cover the coolant mass. In the construction under the invention, no circumferential retainment is prescribed on the fixing of the primary vessel to the roof structure, so that the tangential stresses are suppressed on the edge. The leak tight jacket is not in contact with the coolant and, as it is cooled by a blanket gas virtually in contact with all its internal area, it is not subjected to the high temperature difference between the surface of the mass and the pit roof. Should the need be felt, an upper circumferential area of the leak tight jacket could be internally lined with a lagging to reduce the heat transmission by radiation from the coolant mass and could be cooled from the outside. In a preferred construction, the ties include rods or straps articulated or welded to an upturned skirt secured inside the primary vessel at a height situated under the level of coolant mass surface.

- 4429 SEAL FOR TURNABLE LIDS ON NUCLEAR REACTORS. Jansing, W.; Roehrs, H.; Rothfuss, H. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). German (FRG) Patent 1,921,377/C/. 4 Aug 1977. 4p. (In German).

This seal guarantees greatest leak-proofness even when turning the turn-lid and keeps off abrasion and dirt from the interior of the reactor vessel. This is caused by various individual seals in the space above the horizontal flange which closes the vessel at the top. An outer ring is removably supported on an outer diameter of this flange with the narrow side of its polygonal cross-section. An inner ring turns within this body with its narrow side on spheres of a support. The turn-lid is centred with its largest diameter in this inner ring and is supported by a part of the same body protruding it by means of screws. The vertical ring gap between the two ring bodies is sealed by two inflatable hollow seals; they are coated with PTE.

- 4430 (JAPFNR--264) FBR STRUCTURAL MATERIAL TEST PROGRAM. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Aug 1977. Translation of N--251-77-22 39p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

In parallel with development of FBR, Power Reactor and Nuclear Fuel Development Corporation (PNC) has planned and carried out a series of structural material tests in order to establish the rational temperature designs for structural components. These material tests are partially conducted at the O-arai Engineering Center (PNC) but mostly undertaken at the plant fabricator, material manufacturers, academic societies and associations, national research institutes and universities at the request of PNC. Work to date and future plans are discussed.

- 4431 THEORETICAL PREDICTION OF THE THERMAL CONDUCTIVITY OF URANIUM CARBIDE VAPOR. Cox, D.L.; Cronenberg, A.W. (New Mexico Univ., Albuquerque (USA)). J. Nucl. Mater.; 67: No. 3, 326-331(Aug 1977).

Theoretical predictions are made for the thermal conductivity of uranium carbide vapor. The calculations take into account collisional, dissociation, and chemical reaction effects.

- 4432 THEORETICAL PREDICTION OF THE THERMAL CONDUCTIVITY OF URANIUM NITRIDE VAPOR. Kullasha, M.A.; Cronenberg, A.W. (New Mexico Univ., Albuquerque (USA)). J. Nucl. Mater.; 67: No. 3, 320-325(Aug 1977).

A theoretical calculation of the thermal conductivity of uranium nitride vapor has been carried out. The calculational procedure is based on consideration of collisional, polyatomic, dissociation and reaction effects and only effects which do not perturb values by more than a factor of two are considered. Parameters such as the thermal conductivity of vapors are necessary for safety analyses of potential nuclear fuels for LMFBR reactors, such as uranium nitride.

- 4433 TAIL END OF THE FUEL CYCLE: COPING WITH THE CARTER EDICT. Olds, F.C. Power Eng.; 81: No. 8, 38-46(Aug 1977).

President Carter's unilateral attempt to achieve a worldwide moratorium on reprocessing, recycle, and breeder development has had a cold reception. The U.S. alone appears to honor the moratorium, and its prestige suffers as the bill of particulars against the plan takes form. Isotopic denaturing of plutonium fuel, however, is a promising new means for rendering the fuel cycle highly diversion-resistant. In this process the fissile isotopes of plutonium would be tied almost inseparably into a chain of fertile isotopes. This renders the material unsuitable for weapons and difficult to handle outside of the fuel cycle. For isotopic denaturing to be most effective, it should be instituted as widely as possible, throughout the world. This requires a number of procedural, political, and international policy measures. In short, the nuclear power nations need to agree on the use of denatured fuel, and to restrict export reactors to those using denatured fuel.

- 4434 DIAGNOSING MALFUNCTIONS IN LIQUID SODIUM HYDROSTATIC BEARINGS. Wilcock, D.F. (Mech Technol Inc, Latham, NY). Lubr. Eng.; 33: No. 8, 412-416(Aug 1977).

One of the principal methods being developed for the fast breeder reactor power plant for the transfer of heat from the reactor core is the use of a liquid sodium transfer loop. Operation of this loop requires large pumps capable of handling liquid sodium at temperatures over 1000 F. The lower impeller bearing in these vertical shaft pumps is conventionally a multipocket hydrostatic bearing supplied with pressure from the pump discharge. Continued operation of this bearing is vital to the power plant operation. Hence, diagnosis of its condition is important to assessing plant condition and scheduling outages for repair. A logical analysis of hydrostatic bearing faults and the observable symptoms which may be displayed as the result of each fault can be used to establish a desirable level of instrumentation. If proximeters, pressure sensors, and a bearing supply flow meter can be used, each type of fault that has been identified can be distinguished clearly. With this level of instrumentation, the power plant operator can

reach rational decisions related to continued operation of a large liquid metal pump after a bearing malfunction has been signalled. Furthermore, the fault-symptom logic forms a natural basis for automation of the instrumented surveillance of large sodium pumps.

- 4435 EFFECT OF PHASE INSTABILITY OF STAINLESS STEEL ON THE CLADDING DEFORMATION PROFILES OF LMFBR FUEL ELEMENTS. Hofman, G.L.; Walters, L.C.; McVay, G.L. (Argonne National Lab., Ill. (USA)). J. Nucl. Mater.; 67: No. 3, 289-294(Aug 1977).

Enhanced deformation rates in LMFBR fuel element cladding are shown to be consistent with the occurrence of carbide precipitates. Swelling and creep equations, when modified to include the effects of carbide precipitation, account for such phenomena as the second diameter peak and certain large heat-to-heat variations observed in the cladding and irradiated elements.

- 4436 EFFECTIVE TEMPERATURE OF AN EXPANDING TUBE WITH A RADIAL TEMPERATURE GRADIENT OVER THE WALL. Van Witzenburg, W.; Janssen, L.G.J.; Prijs, J. (Energieonderzoek Centrum Nederland, Petten). Nucl. Technol.; 35: No. 1, 184-187(Aug 1977).

By using the finite-strain theory, an effective temperature, defined as a uniform temperature producing the same deformation rates as the actual temperature gradient, of a thick-walled internally pressurized closed-end tube with an over-the-wall radial temperature gradient, is expressed in terms of original tube geometry, strain, temperature distribution, and creep properties of the tube material. For a stainless-steel cladding tube of a fission reactor fuel pin, with a radial temperature gradient of 50°C over the wall, the effective temperature equals the average of the instantaneous inner and outer surface temperatures to within 10°C.

- 4437 IMPORTANCE OF HEAT TRANSFER IN HYPOTHETICAL CORE DISRUPTIVE ACCIDENT ANALYSIS. Abramson, P.B. (Argonne National Lab., IL). Nucl. Technol.; 35: No. 1, 87-96(Aug 1977).

In the very unlikely event of a loss-of-flow accident in a liquid-metal fast breeder reactor being accompanied by complete failure to scram, the reactor could go prompt critical, generating a large amount of neutronic heat on a millisecond time scale. We find that fuel-to-steel heat transfer has a minimal influence upon the neutronic energy deposition during the prompt burst but that it can play an important role in material behavior in later stages of the hypothetical core disruptive accident. Furthermore, results obtained indicate that calculations of thermodynamic potential energy through adiabatic expansion to one atmosphere are conservative if performed at the end of the prompt burst and that fuel-to-steel heat transfer may significantly reduce the available work energy within the next 20 ms.

- 4438 CORROSION OF STEELS IN MOLTEN SODIUM HYDROXIDE. Newman, R.N.; Smith, C.A.; Smith, R.J. (Berkeley Nuclear Labs., Eng.). J. Electrochem. Soc.; 124: No. 8, 1247-1252(Aug 1977).

The role of sodium hydroxide corrosion is discussed in relation to the wastage of materials observed in fast reactor boilers under fault conditions in the vicinity of a water leak into sodium. An experimental technique to study the corrosion under varying

conditions is described. The results presented are for $2\frac{1}{4}$ Cr-1Mo obtained in static sodium hydroxide in a closed volume over the temperature range 1033°--1273°K. It is found that the corrosion rate can be followed by monitoring the hydrogen produced by the reaction, which can be written as $\text{Fe} + 2\text{NaOH} = \text{NaFeO}_2 + \text{NaH} + \frac{1}{2} \text{H}_2$. After an initial acceleration period the rate law is parabolic. The effect on the corrosion rate of melt and cover gas composition has been in part investigated, and the relevance of mass flow of reactant is discussed.

- 4439 ACOUSTIC DETECTION FOR SMALL-LEAK SODIUM-WATER REACTION. Nei, H.; Ohshima, I.; Ujihara, K.; Hori, M. (O-arai Eng Cent, Power React and Nucl Fuel Dev Cop, Ibaraki-ken, Jpn). J. Nucl. Sci. Technol. (Tokyo); 14: No. 8, 558-5264 (Aug 1977).

Characteristics of acoustic signals produced by sodium-water reaction due to steam injection and by Ar gas injection into sodium were experimentally investigated. Acoustic signals were measured by using Kistler 808A and 815A5 accelerometers. Root mean square (RMS) measurements and frequency analysis of the signal were conducted. The RMS measurements could detect a small water leakage into sodium, as small as 0.07 g/sec, in the present loop. The peaks in a frequency spectrum were caused by the natural vibration of a rod on which the acoustic transducer was mounted. The RMS was approximately proportional to the one-third power of the steam leak rate and increased to some extent with the ambient sodium temperature. RMS values, both for sodium-water reaction and Ar gas injection, were about the same order of magnitude, when the data were plotted against the volumetric flow rates of steam and Ar gas. 11 refs.

- 4440 CONSEQUENCES OF THE OVG MUENSTER'S DECISION ON KALKAR. Klein, H.H. (Deutscher Bundestag, Bonn (Germany, F.R.)). Ausschuss fuer Wirtschaft). Dtschl.-Union-Dienst; 31: No. 166, 1-3 (Aug 1977). (In German).

In the authors opinion, the higher administrative court at Muenster was right constitutionally as well as politically. Constitutionally: The Federal Constitutional Court has since the medical society verdict of 1972- often and with emphasis expressed itself in the sense that the greater the relevance of a decision to basic rights, and the more far-reaching its significance for everybody, all the more stringent the legislator's guidance for the executive has to be. Politically: The fact, that parliament has withdrawn itself concerning State guidance in the sense of political pilotage in favour of the government, apart from not being very effective, threatens to become a thorn in the side of many quarters. The article briefly outlines in which form parliament ought to be involved.

- 4441 ACOUSTIC DETECTION FOR SMALL-LEAK SODIUM-WATER REACTION. Nei, H.; Ohshima, I.; Ujihara, K.; Hori, M. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). J. Nucl. Sci. Technol. (Tokyo); 14: No. 8, 558-564 (Aug 1977).

Characteristics of acoustic signal produced by sodium-water reaction due to steam injection and by Ar gas injection into sodium were experimentally investigated. Acoustic signal was measured by using Kistler 808A and 815A5 accelerometers. Root mean square (RMS) measurements and frequency analysis of the signal were conducted. The RMS measurements

could detect a small water leakage into sodium, as small as 0.07g/sec, in the present loop. The peaks in a frequency spectrum were caused by the natural vibration of a rod on which the acoustic transducer was mounted. The RMS was approximately proportional to the one-third power of the steam leak rate and increased to some extent with the ambient sodium temperature. RMS values, both for sodium-water reaction and Ar gas injection, were about the same order of magnitude, when the data were plotted against the volumetric flow rates of steam and Ar gas.

- 4442 DECAY HEAT REMOVAL TEST OF SPENT FUEL IN 'MONJU' EX-VESSEL TRANSFER MACHINE. Yukishima, I.; Hayakawa, S.; Ishiguro, J. (Fuji Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). FAPIG (Tokyo); No. 86, 2-7 (Aug 1977). (In Japanese).

Decay heat of spent fuel of the prototype fast breeder reactor 'Monju' during the transit is removed by direct argon gas cooling around and within the spent fuel subassemblies. Heat removal test was conducted to confirm the cooling characteristics with a model of the spent fuel subassembly and an argon gas cooling loop. The specifications of the test facility are heating and cooling capacity of 200 kW, argon gas blower capacity of 2,000 mmAq x 15 m³/min. and the fuel subassembly of approximately actual size. The test conditions are 1 - 8 kW of heat generation, about 150 deg. C of argon inlet temperature, and 0 - 10 m³/min. of argon flow rate. The correlation between Reynolds number and Stanton number or friction coefficient for cooling the spent fuel subassembly and the effect of sodium adhesion were experimented in this test, and these experimental data were compared to the analytical results using HECTIC-2a code for steady state and HECTIC-3 code for unsteady state. These codes are available to calculate accurately the maximum temperature of fuel pins and its time transient. No significant effect of sodium adhesion was observed.

- 4443 SUPPORT STRUCTURE FOR THE LATERAL NEUTRON SHIELD SYSTEM OF A FAST REACTOR CORE. Gallet, B.; Venot, R. (to Commissariat a l'Energie Atomique). US Patent 4,038,136. 26 Jul 1977. Priority date 11 Jun 1974, France, 4p.

A hexagonal ring structure whose internal contour conforms to the external contour of the reactor core rests on the periphery of the reactor diagrid and is constituted by a plurality of layers of metallic plates. The layers are maintained in relative positional relation by means of clamping members, the plates of one layer being angularly displaced with respect to the plates of adjacent layers. The bottom end-connectors of the elements constituting the lateral shield system are fitted in vertical through-holes formed in the ring structure.

- 4444 NUCLEAR REACTOR FUEL ELEMENTS. Linning, D.L.; Dodd, J.A. (to United Kingdom Atomic Energy Authority). US Patent 4,038,138. 26 Jul 1977. Priority date 7 Mar 1974, United Kingdom of Great Britain and Northern Ireland (UK), 2p.

A nuclear reactor fuel element is described which has wire wrap spacing means of a form which contacts the fuel element at a series of regularly spaced intermittent points so that coolant can flow between the wire wrap and the fuel element and between the points of contact. The wire wraps may be of single wire coiled coil form or twin interwound wires. 2 figures.

4445 NUCLEAR REACTOR FUELING SYSTEM. Wade, E.E. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). French Patent 2,359,485/A/. 20 Jul 1977. Priority date 22 Jul 1976, United States of America (USA), 17p. (In French).

Nuclear fuel transfer gear to move nuclear fuel assemblies between a storage vessel and a nuclear reactor vessel placed against this storage vessel, each of these vessels having coolant inlets and outputs for the coolant to flow through them. This gear includes: - a connexion system forming a passage between the reactor vessel and the storage vessel located below the level of the coolant, so as to permit the transfer of the fuel assemblies between the reactor vessel and the storage vessel whilst maintaining these fuel assemblies completely immersed in the coolant; - a first system for the transfer of fuel assemblies fitted in the reactor vessel above the reactor core; - and a second fuel assembly transfer system fitted in the storage vessel above the fuel assembly storage area. These facilities are characterised in that the first and second fuel assembly transfer systems include a first and second respective revolving appliance, to hold the fuel assemblies and to align them with the connexion system so as to transfer them through the passage formed by this connexion system.

4446 NUCLEAR REACTOR FUEL ELEMENT ASSEMBLIES. Betts, C.; Collinson, A.E.; Hart, D.F. (to United Kingdom Atomic Energy Authority). US Patent 4,036,690. 19 Jul 1977. Priority date 31 Dec 1974, United Kingdom of Great Britain and Northern Ireland (UK), 4p.

A nuclear reactor fuel element assembly comprising a bundle of fuel elements in a tubular wrapper for conducting liquid metal coolant flow has a cavitation suppressing impedance. The impedance comprises a tubular member housing a series of flow defining plates, the plates comprising two sets arranged so that plates of each set are positioned alternately along the tubular member, the plates of one set being of woven wire mesh and those of the other set being thick apertured plates.

4447 HEAT EXCHANGER. (to Combustion Engineering, Inc., Windsor, Conn. (USA)). Netherlands Patent 7,707,862/A/. 14 Jul 1977. 9p. (In Dutch).

A liquid metal heated tube and shell heat exchanger where straight tubes extend between upper and lower tube sheets. In order to prevent thermal stress problems, one tube sheet is fixed to the shell, and the other tube sheet is sealed to the shell by means of a flexible bellows. In the event of a catastrophic bellows failure, a housing that utilizes a packing gland sliding seal is used to enclose and back-up the bellows. Also, a key and slot arrangement is provided for preventing relative rotation between the shell and tube sheet which could damage the bellows and cause failure thereof. This exchanger is seen to be of use in sodium cooled reactors between the liquid sodium circuit and the steam generator.

4448 DISPERSION NUCLEAR FUEL AND PROCESS FOR ITS MANUFACTURE. Ondracek, G.; Nazare, S.; Thummmler, F. (to Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). German(FRG) Patent 1,916,370/B/. 14 Jul 1977. 6p. (In German).

A dispersion nuclear fuel for fast reactors is described, whose reprocessing is possible.

It consists of a heat conducting metal matrix (pure copper or copper alloy) which hardly absorbs any fast neutrons and a compatible fuel or breeding material for this matrix, such as U_2O or PuO_2 . The manufacturing process is described. The copper alloy contains at least one element from the group nickel, aluminium or chromium.

4449 NUCLEAR FUEL ASSEMBLY. Betten, P.R. (to Combustion Engineering, Inc., Windsor, CT (USA); Deutsches Patentamt, Muenchen (Germany, F.R.)). German(FRG) Patent 2,653,250/A/. 14 Jul 1977. 16p. (In German).

This is a nuclear fuel assembly for a liquid metal cooled reactor with improved coolant distribution in the peripheral and inner coolant flow channels. Swelling and radiation-induced growth are also taken into account. The parallel fuel rods each have a spirally wound spacer on the surface. A bundle of them is inserted in a shell, which has projections on the inside. These have a pattern, which is given by the angle of the spiral on the fuel rods. The play in the peripheral rods is reduced by this and the coolant flow in the inner-channels is improved. Compared to the outer coolant section of hexagonal cross section, the shell also has projections on its surface, which act as springs and are made in the form of ribs.

4450 NUCLEAR REACTOR FUEL ELEMENT SUB-ASSEMBLIES. Waite, E.; Ford, J. (to UKAEA). British Patent 1,478,985/A/. 6 Jul 1977. 7p.

Reference is made to fuel element assemblies for fast reactors. Normally these consist of a number of upstanding parallel pins between which liquid metal coolant flows. To facilitate insertion and withdrawal of fuel the pins are grouped in side-by-side sub-assemblies each comprising a bundle of pins enclosed by a tubular wrapper. The pins are end supported on a lower grid within the wrapper and are free to move longitudinally within a clearance limit afforded by an upper grid within the wrapper. The pins must be free to move longitudinally under irradiation growth and linear thermal expansion, but for stability of the reactor core it is desirable that the pins be held captive. Captivating the pins has been found difficult because of their close spacing and the need to maintain adequate coolant flow passages. In the arrangement described the pins have peripherally enlarged heads at one end and the heads are held captive between a fixed end abutment grid and a fixed cellular grid, with provision of coolant flow passages. The pins may have their enlarged heads at the ends which are intended to be uppermost in a reactor and be suspended from the cellular grid. Alternatively the pins may have their enlarged heads at the ends intended to be lowermost, in which case the pins are supported by the end abutment grid.

4451 METHOD OF DETECTING LEAKAGE OF REACTOR CORE COMPONENTS OF LIQUID METAL COOLED FAST REACTORS. Holt, F.E.; Cash, R.J.; Schenter, R.E. (to Energy Research and Development Administration). US Patent 4,033,813. 5 Jul 1977. Filed date 22 Oct 1975. 4p. PAT-APPL-624,804.

Disclosed is a method of detecting the failure of a sealed nonfueled core component of a liquid-metal cooled fast reactor having an inert cover gas. A gas mixture is incorporated in the component which includes Xenon-124; under neutron irradiation, Xenon-124 is converted to radioactive Xenon-125. The cover

gas is scanned by a radiation detector. The occurrence of 188 Kev gamma radiation and/or other identifying gamma radiation-energy level indicates the presence of Xenon-125 and therefore leakage of a component. Similarly, Xe-126, which transmutes to Xe-127 and Kr-84, which produces Kr-85/sup m/ can be used for detection of leakage. Different components are charged with mixtures including different ratios of isotopes other than Xenon-124. On detection of the identifying radiation, the cover gas is subjected to mass spectroscopic analysis to locate the leaking component. 2 claims.

- 4452 FUSION MATERIALS. Rovner, L.H. (Gen At Co). Power Eng.; 81: No. 7, 61-65(Jul 1977).

A study is made of materials requirements for tokamak nuclear fusion systems. Particular attention is paid to the problem of material erosion from the first wall due to the impingement of high-energy ions from the plasma. It is therefore suggested that a low atomic number material such as carbon or silicon carbide be chosen to interface with the plasma. Other problems discussed include irradiation-induced swelling of material and the use of graphite to moderate the damage. Finally, the problem of radiation-induced resistivity increases in superconducting magnets is discussed.

- 4453 HIGHER ADMINISTRATIVE COURT AT MUENSTER EXPRESSES DOUBTS CONCERNING THE CONSTITUTIONAL CHARACTER OF SECTION 7 OF THE ATOMIC ENERGY ACT. Gut; No. 19, 1-2(Jul 1977). (In German).

In connection with the procedure in contentious matters over the SNR 300, the chairman of the 7th senate of the Higher Administrative Court at Muenster has sent participants in the procedure a document, in which it is pointed out that in order to come to a decision in the law case in question within the framework of dutiful exercise of the judicial right of control, the constitutional character of section 7 of the Atomic Energy Act would have to be examined and also, whether the principles of basic law would be still warranted if the executive, and not parliament, makes decisions with considerable consequences (ultimate storage, plutonium industry, etc.) - Note: The matter was passed on to the Federal Constitutional Court on the 18th August 1977 in order that the constitutional character of section 7 of the Atomic Energy Act may be examined.

- 4454 BEHAVIOR OF STEEL X6CRN1811 UNDER COMBINED CREEP AND CYCLIC LOADING AT 823°K. Breitling, H.; Grosser, E.D.; Lorenz, H. Arch. Eisenhuettenwes.; 48: No. 7, 403-507(Jul 1977). (In German).

The behavior of the austenitic steel X6CrNi18 11 resembling AISI 304 in composition, was subjected to three series of tests at 823 K; viz, cyclic loading followed by creep loading to rupture (I); creep loading followed by cyclic loading to rupture (II); and successive short alternating creep and cycling loading to rupture (III). The results were evaluated using the linear damage rule. The damage factors, D, determined vary between 0.7 and 1.6. The highest values were obtained in series I and the lowest in series III. The lower D values for alternating loading can be attributed to the softening processes taking place when the type of loading is changed. A comparison of the data obtained with the damage factors indicated in the ASME BPVC, CC 1592, shows that

the D factors determined from the experimental results are, as in the ASME BPVC, lowest when the two damage fractions are approximately equal. Most of the D factors lie above the ASME values. The ASME curve represents a conservative lower limit of the results reported. (The steel is to be used in pressure vessel and its equipment and in the heat exchange system of a German fast sodium cooled breeder reactor, SNR 300). 12 refs.

- 4455 POSSIBILITIES FOR SOLVING THE PROBLEM OF REPROCESSING FUEL FROM FAST BREEDERS. SURVEY ON THE BASIS OF REPORTS BY EXPERTS ON THE MEETING OF THE IAEA IN MAY 1976 IN LENINGRAD. Arch. Energiewirtsch.; 31: No. 7, 605-622(Jul 1977). (In German).

From Meeting of the experts' group on reprocessing of fuels from liquid-metal cooled fast breeders; Leningrad, USSR (17 - 21 May 1976).

It is reported on the state of the R+D work on reprocessing of irradiated nuclear fuel from sodium-cooled fast breeders in several countries (France, FRG, Belgium, UK, Italy, Japan, USA). The basic work is concentrated on methods of separating the sodium from the fuel elements, transport, storage, cutting and dissolving of the fuel elements; further on separating insoluble particles from the fuel solution, adapting the Purex process to the higher Pu content and burn-up, removing gaseous fission products, and off-gas cleaning. Russian contributions to this catalogue of subjects are missing.

- 4456 HEAT TRANSFER MEASUREMENTS IN AN AIR MODEL OF A PARTLY BLOCKED REACTOR FUEL SUBASSEMBLY. Judd, A.M.; Maltcotsis, G. (Cambridge Univ. (UK). Dept. of Engineering). J. Br. Nucl. Energy Soc.; 16: No. 3, 237-242(Jul 1977).

The effect of blockages of up to six subchannels on heat transfer from a heated pin in a large-scale model of part of a fast reactor fuel subassembly has been measured, using air as a coolant. The results are compared with those of similar measurements in other, simpler, geometries. It is found that the surface heat transfer coefficient is in most cases increased, but there is a tendency for the increase in certain parts of the flow to be smaller for larger blockages. It is concluded that overheating in the region of the blockage is due to increased coolant temperatures rather than poor surface heat transfer.

- 4457 TURBULENCE MEASUREMENTS IN AN AIR MODEL OF A PARTLY BLOCKED REACTOR FUEL SUBASSEMBLY. Judd, A.M. (Cambridge Univ. (UK). Dept. of Engineering). J. Br. Nucl. Energy Soc.; 16: No. 3, 243-248(Jul 1977).

The Eulerian integral time-scale and the kinetic energy of the turbulence have been measured at various points in the recirculating regions behind blockages of up to 13 subchannels in a large-scale model of part of a fast reactor fuel subassembly using air to represent the coolant. It is found that roughly the time-scale is directly proportional and the kinetic energy is inversely proportional to the diameter of the blockage. From these measurements approximate values of the eddy diffusivity can be deduced, and are found to be independent of the blockage size. In a fast reactor, the eddy diffusivity behind a blockage would be far greater than the conductive thermal diffusivity.

- 4458 STRESS ANALYSIS OF FUEL CLADDINGS WITH AXIAL FINS INCLUDING CREEP EFFECTS. Krieg, R. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). Nucl. Eng. Des.; 42: No. 2, 357-380 (Jul 1977).

For LMFBR fuel claddings with axial fins the stress and strain fields are calculated which may be caused by internal pressure, differential thermal expansion and irradiation induced differential swelling. To provide an appropriate description of the cladding material it is assumed that the total strain is the sum of a linear elastic and a creep term, where the latter one includes the thermal as well as the irradiation induced creep. First the linear elastic problem is treated by a semi-analytical method leading to a bipotential equation for Airys' stress function. Solving this equation analytically means that the field equations valid within the cladding are satisfied exactly. By applying a combined point matching-least square-method the boundary conditions could be satisfied approximately such that in most cases the remaining error is within the uncertainty range of the loading conditions. Then the nonlinear problem which includes creep is approximated by a sequence of linear elastic solutions with time as parameter. The accumulated creep strain is treated here as an imposed strain field. To study the influence of different effects such as fin shape, temperature region, irradiation induced creep and swelling or internal pressure, a total of eleven cases with various parameter variations are investigated. The results are presented graphically in the following forms: stress and strain distributions over the cladding cross section for end of life conditions and boundary stresses and strains versus time.

- 4459 PREDICTION OF TEMPERATURE DISTRIBUTION OF A SUBASSEMBLY INCLUDING INTERSUBASSEMBLY HEAT TRANSFER. Chang, L.K. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 42: No. 2, 223-231 (Jul 1977).

A three-dimensional thermal hydraulic code, CLUSTER, has been developed in order to predict the temperature profile in a cluster of seven subassemblies during steady state conditions. The subassemblies under consideration include both pin-bundle and reflector types of subassemblies. An implicit finite difference representation of the governing energy equation is employed. Good agreement was obtained between the analytical results and the measured data.

- 4460 BREEDER REACTOR DEVELOPMENT: PROGRESS IN EUROPE AND JAPAN, DISCUSSION ON PRINCIPLES IN THE USA. Hueper, R. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Projekt Schneller Brueter). At. Strom; 23: No. 4, 81-85 (Jul 1977). (In German).

15 refs..

The fast breeder reactor development is making progress in the Soviet Union, France, Great Britain, the Federal Republic (together with Belgium and the Netherlands) and Japan. India and Italy are also active in this field, and other countries such as Spain and Brazil are showing their strong interest. At the same time the United States, who initiated this development as early as 30 years ago, tend to slow down their own breeder activities substantially. The author gives a picture for breeder reactor development outside the USA, in particular in the Federal Republic of Germany. He then proceeds to give an outline of the discussion of principle that is going on in the USA.

- 4461 ENERGY FROM NUCLEAR REACTORS. PT. 4. THE FAST BREEDER: BETTER UTILIZATION OF URANIUM. Hospe, J. VDI (Ver. Dtsch. Ing.) Nachr.; 31: No. 27, 4 (Jul 1977). (In German).

Fast breeders operate with fast neutrons. They are the only reactor type which produces by a breeding process more fission materials than it uses. On account of this property they are of great importance for long-term nuclear fuel supply.

- 4462 ENERGY FROM NUCLEAR REACTORS. PT. 5. FAST BREEDER REACTOR SAFETY. Hospe, J. VDI (Ver. Dtsch. Ing.) Nachr.; 31: No. 28, 10 (Jul 1977). (In German).

The author briefly explains the safety technique applied to the fast breeder and compares it with the techniques for other nuclear power plants.

- 4463 SOME PROPERTIES OF IRON-BASE MATERIALS WITH CALCIUM FLUORIDE ADDITIONS FOR FRICTIONAL UNITS OF NUCLEAR PLANTS. Fedorchenko, I.M.; Shevchuk, Yu.F.; Miroshnikov, V.N.; Tuznikov, A.F.; Markov, V.G.; Gorynin, I.V.; Krasnoschekov, M.M.; Mostovskii, D.M. Sov. Powder Metall. Met. Ceram. (Engl. Transl.); 16: No. 7, 544-548 (Jul 1977).

The results of this investigation into the corrosion resistance and diffusion welding of sintered iron-base materials and into the effect of neutron irradiation upon their frictional properties suggest that a material of composition $\text{Fe}_{15}\text{Mo}_6\text{CaF}_2$ should prove useful in frictional units of sodium-cooled nuclear power plants. The material proved to be about twice as resistant to corrosion by liquid sodium as other materials investigated. 7 refs.

- 4464 EFFECT OF CARBIDES ON THE DECARBURIZATION BEHAVIOR OF 2.25 CR-1 MO STEEL IN HIGH TEMPERATURE SODIUM. Shida, Y.; Yoshikawa, K.; Morioishi, T. (Sumitomo Metal Industries Ltd., Amagasaki, Hyogo (Japan). Central Research Labs.). J. Jpn. Inst. Met.; 41: No. 7, 697-705 (Jul 1977). (In Japanese).

2 1/4 Cr-1 Mo steel is highly promising for construction materials of the LMFBR secondary system. The effect of carbides on the decarburization rate of 2 1/4 Cr-1 Mo steel in high-temperature sodium was studied. The effect was investigated by immersing specimens with different metallurgical structures and various kinds of carbides in static liquid sodium at temperatures from 550 to 700°C. The changes of carbides in the composition, phase and morphology were examined by the X-ray diffraction method, chemical analysis of carbides, electromicroscopic observation by extraction replica and the electron diffraction method. The following results were obtained. (1) Decarburization rate of 2 1/4 Cr-1 Mo steel in sodium can be related to phase, morphology, transformation rate and the coagulation process of carbide precipitates. Carbides dissolve in the order of $\text{M}_3\text{C} > \text{lamellar } \text{M}_{23}\text{C}_6 > \text{granular } \text{M}_7\text{C}_3$ asymptotically equals granular M_{23}C_6 asymptotically equals $\text{M}_2\text{C} > \text{M}_6\text{C}$. (2) Carbides dissolve so as to spheroidize themselves. Diffusion of Cr and Mo from carbide to the matrix seems to control the dissolution rate of comparatively stable carbide. (3) Carbides should be stabilized in order to minimize the rate of decarburization. It is most appropriate to make M_6C a dominant carbide phase in steel.

- 4465 CIRCUMFERENTIAL NONUNIFORMITY OF CLADDING RADIATION SWELLING OF FAST REACTOR

PERIPHERAL FUEL ELEMENTS. Reutov, V.F.; Parkhutdinov, K.G. At. Energ. (USSR); 43: No. 1, 12-15(Jul 1977). (In Russian).

The results are presented of the investigation into the perimeter radiation swelling of Kh18N10T stainless steel cladding in different cross sections of a peripheral fuel element of the BR-5 reactor. The fluence on the cladding is $1.8-2.9 \times 10^{22}$ fast neutr/cm², the operating temperatures in different parts of the fuel element being 430 deg to 585 deg C. There has been observed circumferential non-uniformity of the distribution, concentration, and of the total volume of radiation cavities, which is due to temperature non-uniformity along the cladding perimeter. It is shown that such non-uniformity of radiation swelling of the cladding material may result in bending of the peripheral fuel element with regard to the fuel assembly sheath walls.

4466 FISSION PRODUCT YIELDS DURING IRRADIATION OF ²³⁵U AND ²³⁹PU IN THE BOR-60 REACTOR. Gabeskiriya, V.Ya.; Gryzina, V.V.; Nivokov, V.V.; Prokopenko, V.S.; Prokop'ev, V.M.; Tikhomirov, V.V.; Chetverikov, V.A. At. Energ. (USSR); 43: No. 1, 59-60(Jul 1977). (In Russian).

BOR-60 REACTOR; DATA; ERRORS; FISSION PRODUCTS; FISSION YIELD; MASS NUMBER; PLUTONIUM 239; URANIUM 235; REACTOR LATTICE PARAMETERS

4467 BUILD-UP AND DECAY OF ACTINIDE NUCLIDES IN FUEL CYCLE OF NUCLEAR REACTORS. Tasaka, K.; Kikuchi, Y.; Shindoh, R.; Yoshida, H. (Japan Atomic Energy Research Inst., Tokai, Ibaraki. Tokai Research Establishment). J. Nucl. Sci. Technol. (Tokyo); 14: No. 7, 519-531(Jul 1977).

In order to assess the feasibility of utilizing plutonium in thermal reactors, build-up and decay of actinide nuclides have been studied for BWR, PWR, HWR, HTGR and LMFR, which are uranium-oxide fueled or mixed-oxide fueled, and which produce electric power of 1,000 MW. The following items were examined: (1) quantities of actinide nuclides build-up in the reactor, (2) build-up and decay of activities of actinides in the spent fuel, (3) build-up and decay of activities of actinides after reprocessing, and (4) variation of isotopic composition of plutonium with high burn-up. It is concluded from the calculated results that precautions should be taken against high activities of resultant actinides if plutonium is utilized as a fissile material for thermal reactors. To make reprocessing and high-level waste management easy and practical, it is recommended that a thermal reactor should be fueled with uranium, the plutonium produced in a thermal reactor should be used in a fast reactor, and plutonium produced in the blanket of a fast reactor is more appropriate as fast reactor fuel than that from a thermal reactor.

4468 STATISTICAL ERROR OF CALCULATING REACTIVITY BALANCE. Kebabze, B.V.; Efimov, V.N.; Adamovskii, L.A. At. Energ. (USSR); 43: No. 1, 36-37(Jul 1977). (In Russian).

BACKGROUND NOISE; BOR-60 REACTOR; CORRELATION FUNCTIONS; ERRORS; FLOWSHEETS; NUMERICAL SOLUTION; REACTIVITY; REACTOR NOISE; TIME DEPENDENCE

4469 FAST BREEDER NUCLEAR REACTOR COOLED WITH LIQUID METAL. Ackroyd, R.T.; Perks, M.A. (to UKAEA London Office). German(FRG) Patent 1,957,090/B/. 30 Jun 1977. 6p. (In German).

A variation of the construction of a liquid metal cooled nuclear reactor (a fast breeder)

is proposed, which has the following zones, starting on the inside and working outwards: A first breeding zone, which is surrounded by a nuclear fuel zone, a second breeding zone with surrounding nuclear fuel and a third breeding zone, which reaches to the outside of the reactor core. One obtains a high breeding yield with short doubling time. Various geometrically different variants are described.

4470 PROTECTIVE COATING FOR FLOOR COOLER. Friedrich, H.J.; Blokker, E. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). German(FRG) Patent 2,557,884/A/. 30 Jun 1977. 13p. (In German).

This protective layer (consisting of two layers) takes the molten fuel in a hypothetical reactor accident. The metallic circular floor trough contains the two layers of different thicknesses according to the invention. The upper layer, 5 to 10 cm thick, of uranium dioxide or (for sodium cooled fast breeder reactors) of thorium 'bodies' of high melting point and low conduction produces widespread dispersion of the liquid fuel with slow penetration. It prevents the formation of a core physical critical mass and does not react, or reacts only slightly, with the liquid fuel. The lower layer consisting of preferably hexagonal closely packed 15 to 20 cm high graphite bodies, conducts the heat flow from above more quickly than it arrives, into the floor trough. The melting point of the lower material in the separating layer from the upper material is not exceeded.

4471 SENSOR FOR IONIZABLE ELEMENTS. Berkey, E.; Reed, W.A. III; Hickam, W.M. (to Westinghouse Electric Corp., Pittsburgh, PA (USA); Deutsches Patentamt, Muenchen (Germany, F.R.)). German(FRG) Patent 2,557,132/A/. 23 Jun 1977. 13p. (In German).

A sensor is described which is to detect thermally ionizable elements or molecules in air, water vapour or oxygen or to be used as alkali leak detector in vacuum systems, e.g. in the pipe system of a liquid-metal cooled FBR. The sensor consists of an filament made of thorium-containing iridium as cathode with a temperature upto 1000°C and an anode sheet of molybdenum, nickel or stainless steel.

4472 PUMP TANK DIVIDER PLATE FOR SUMP SUCTION SODIUM PUMPS. George, J.A.; Nixon, D.R. (to Energy Research and Development Administration). US Patent 4,029,123. 14 Jun 1977. Filed date 20 Aug 1975. 6p. PAT-APPL-606,043.

A circular plate extends across the diameter of a sump suction pump, with a close clearance between the edge of the plate and the wall of the pump tank. The plate is located above the pump impeller, inlet and outlet flow nozzles but below the sodium free surface and effectively divides the pump tank into two separate chambers. On change of pump speed, the close fitting flow restriction plate limits the rate of flow into or out of the upper chamber, thereby minimizing the rate of level change in the tank and permitting time for the pump cover gas pressure to be varied to maintain an essentially constant level. 4 claims, 3 figures.

4473 TUBULAR GAGE FOR A LIQUID-METAL-COOLED FAST BREEDER REACTOR. Hutter, E.; Tuma, L.A. (to Energy Research and Development Administration). US Patent 4,028,811. 14 Jun 1977. Filed date 2 Mar 1976. 12p.

PAT-APPL-663,179.

Spring-loaded plungers are arranged about a housing for insertion into a polygonal tube, one plunger for each side of the tube. Each plunger has a locking cam and sliding wedge mechanism which can overcome the spring force associated with the plunger and lock it in any position. The wedges are operated by a rod movable axially in the housing. Several housings with their associated plungers can be stacked. The stack is lowered into the polygonal tube with all of the plungers locked in a fully inward position. When the stack is in the tube, each wedge is moved to release its locking cam, allowing each of the plungers to spring outward against an inner side of the tube. Each housing will thus gage the internal dimensions of the tube at its elevation. The plungers are locked in position, the entire stack is rotated to bring the plungers into the corners described by the intersections of the flat sides, and the stack is removed from the tube whereupon the dimensions across opposite locked plungers may be read by a micrometer.

- 4474 PRESENT STATE AND DEVELOPMENT POTENTIAL OF HIGH TEMPERATURE REACTORS. Hardt, L. Glueckauf; 113: No. 11, 545-551(2 Jun 1977). (In German).

General review of nuclear reactors is followed by evaluation of boiling water, pressurized water, and fast breeder reactors. Characteristics and safety aspects of high temperature reactors are discussed and use of process heat derived from them for coal gasification is considered. Development of reactor with helium turbine is noted. 9 refs.

- 4475 ROLE OF SIMULATION IN THE U.S. LMFBR PROGRAM. Laidler, J.J. (Westinghouse Hanford, Richland, WA). Trans. Am. Nucl. Soc.; 26: 186-187(Jun 1977).

From Transactions of the American Nuclear Society 1977 annual meeting; New York, NY, USA (12 Jun 1977).

LMFBR TYPE REACTORS; REACTOR MATERIALS; PHYSICAL RADIATION EFFECTS; SIMULATION; PARTICLE BEAMS; CHARGED PARTICLES

- 4476 EFFECT OF SWELLING AND CREEP ON LMFBR PERFORMANCE. Baarthold, W.P. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 26: 206(Jun 1977).

From Transactions of the American Nuclear Society 1977 annual meeting; New York, NY, USA (12 Jun 1977).

LMFBR TYPE REACTORS; REACTOR COMPONENTS; PHYSICAL RADIATION EFFECTS; SWELLING; CREEP; PERFORMANCE

- 4477 MATERIALS PROPERTIES UTILIZATION IN A CUMULATIVE MECHANICAL DAMAGE FUNCTION FOR LMFBR FUEL PIN FAILURE ANALYSIS. Jacobs, D.C. (Westinghouse Advanced Reactors Div., Madison, PA). Trans. Am. Nucl. Soc.; 26: 209(Jun 1977).

From Transactions of the American Nuclear Society 1977 annual meeting; New York, NY, USA (12 Jun 1977).

LMFBR TYPE REACTORS; FUEL ELEMENT FAILURE; FUEL CANS; MECHANICAL PROPERTIES; THERMAL STRESSES; PHYSICAL RADIATION EFFECTS

- 4478 STRESS-FREE SWELLING IN TYPE 316 STAINLESS STEEL AT HIGH FLUENCES. Kenfield, T.A.; Appleby, W.K.; Busboom, H.J. (General Electric Co., Sunnyvale, CA). Trans. Am. Nucl. Soc.; 26: 210(Jun 1977).

From Transactions of the American Nuclear

Society 1977 annual meeting; New York, NY, USA (12 Jun 1977).

LMFBR TYPE REACTORS; FUEL CANS; STAINLESS STEEL-316; PHYSICAL RADIATION EFFECTS; SWELLING; NEUTRON FLUENCE

- 4479 EFFECT OF HEAT TREATMENT ON TENSILE PROPERTIES OF ANNEALED 2 1/4 CR-1 MO STEEL. Kleuh, R.L. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 26: 211-212(Jun 1977).

From Transactions of the American Nuclear Society 1977 annual meeting; New York, NY, USA (12 Jun 1977).

CLINCH RIVER BREEDER REACTOR; STEAM GENERATORS; CHROMIUM-MOLYBDENUM STEELS; HEAT TREATMENTS; TENSILE PROPERTIES; MICROSTRUCTURE

- 4480 REACTION RATE OF SODIUM WITH ETHANOL AND OTHER ALCOHOLS. Welch, F.H.; Steele, O.P. III. (Atomics International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 26: 214(Jun 1977).

From Transactions of the American Nuclear Society 1977 annual meeting; New York, NY, USA (12 Jun 1977).

LMFBR TYPE REACTORS; REACTOR COMPONENTS; CLEANING; SODIUM; ETHANOL; ALCOHOLS; CHEMICAL REACTION KINETICS; REMOVAL

- 4481 SIMPLIFIED THERMAL TRANSIENT AND INELASTIC ANALYSIS SCREENING METHODS PER APPENDIX T, CODE CASE 1592 FOR 2 1/4 CR-1 MO STEEL. Veljovich, W.; Skopp, G.; Anderson, T.L. (Atomics International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 26: 214-215(Jun 1977).

From Transactions of the American Nuclear Society 1977 annual meeting; New York, NY, USA (12 Jun 1977).

CLINCH RIVER BREEDER REACTOR; STEAM GENERATORS; CHROMIUM-MOLYBDENUM STEELS; STRESS ANALYSIS; MECHANICAL PROPERTIES; TRANSIENTS

- 4482 CRACKED HEXAGONAL DUCTS. Petroski, H.J. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 26: 215(Jun 1977).

From Transactions of the American Nuclear Society 1977 annual meeting; New York, NY, USA (12 Jun 1977).

LMFBR TYPE REACTORS; FUEL ELEMENT CLUSTERS; FUEL CANS; BRITTLENESS; FRACTURE PROPERTIES; STRESS ANALYSIS; CRACKS; DUCTS

- 4483 TRITIUM PERMEATION THROUGH STEAM GENERATOR MATERIALS. Renner, T.A.; Raue, D.; McPheeters, C.C. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 26: 216(Jun 1977).

From Transactions of the American Nuclear Society 1977 annual meeting; New York, NY, USA (12 Jun 1977).

LMFBR TYPE REACTORS; STEAM GENERATORS; CHROMIUM-MOLYBDENUM STEELS; TRITIUM; PERMEABILITY; NICKEL; CROLOY

- 4484 ENRICHMENT AND CONTROL OPTIMIZATION IN LMFBRs LOSING REACTIVITY DURING BURNUP. Tzanos, C.P. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 26: 232-233(Jun 1977).

From Transactions of the American Nuclear Society 1977 annual meeting; New York, NY, USA (12 Jun 1977).

LMFBR TYPE REACTORS; FUEL MANAGEMENT; BURNUP; REACTOR LATTICE PARAMETERS; OPTIMIZATION

- 4485 DIRECTIONAL INTEGRATION METHOD FOR TWO-DIMENSIONAL FLUID TRANSIENTS. Shin, Y.W. (Argonne National Lab., IL). Trans. Am. Nucl.

- Soc.; 26: 241-242(Jun 1977).
From Transactions of the American Nuclear Society 1977 annual meeting; New York, NY, USA (12 Jun 1977).
LMFBR TYPE REACTORS;SECONDARY COOLANT CIRCUITS;HYDRAULICS;SODIUM;WATER;TWO-DIMENSIONAL CALCULATIONS;TRANSIENTS
- 4486 US UTILITIES ARGUE FOR THE BREEDER AND REPROCESSING. Knox, R. Nucl. Eng. Int.; 22: No. 258, 29-30(Jun 1977).
The 1977 American Power Conference provided an immediate opportunity for electric utilities to make known their reactions to President Carter's nuclear policy statements. Continued development of the fast reactor, Pu cycle and reprocessing were all considered essential. A nuclear power reactor was the most difficult way of producing plutonium for bombs.
- 4487 ANALYSIS OF TRANSIENT FLUID-STRUCTURE PROBLEMS USING AN ADVANCED EULERIAN METHOD. Wang, C.Y. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 26: 239-241(Jun 1977).
From Transactions of the American Nuclear Society 1977 annual meeting; New York, NY, USA (12 Jun 1977).
LMFBR TYPE REACTORS;REACTOR CORE DISRUPTION; HYDRODYNAMICS;MATHEMATICAL MODELS;REACTOR INTERNALS;REACTOR VESSELS
- 4488 SUBCRITICALITY DESIGN CONSIDERATIONS FOR LMFBR SPENT FUEL SHIPPING CASKS. Philbin, J.S.; Dupree, S.A. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 26: 244-245(Jun 1977).
From Transactions of the American Nuclear Society 1977 annual meeting; New York, NY, USA (12 Jun 1977).
LMFBR TYPE REACTORS;SPENT FUEL CASKS; CRITICALITY;SAFETY;DESIGN;SPENT FUELS;TRANSPORT
- 4489 DESIGN OF CONTAINERS FOR SHIPPING UNIRRADIATED LMFBR ASSEMBLIES. Langhaar, J.W. (E. I. duPont de Nemours and Co., Aiken, SC). Trans. Am. Nucl. Soc.; 26: 247(Jun 1977).
From Transactions of the American Nuclear Society 1977 annual meeting; New York, NY, USA (12 Jun 1977).
LMFBR TYPE REACTORS;FUEL ELEMENTS;DESIGN; TRANSPORT;CASKS
- 4490 ENERGY GAP AND THE FAST REACTOR. Hill, J. (UKAEA). Atom (London); No. 248, 106-109(Jun 1977).
The author discusses the case for nuclear energy and in particular for the fast reactor. The real limitations to the fast reactor system are considered from the viewpoints of physics, sodium coolant, fuel, and engineering.
- 4491 REVIEW OF THE POTENTIAL FOR THE USE OF NUCLEAR REACTORS FOR HYDROGEN PRODUCTION. Worley, N.G. (Babcock and Wilcox Ltd., London (UK)). Energy Dig. (London); 6: No. 3, 40-44(Jun 1977).
The possibilities of using reactors to carry out high temperature chemical processes are considered briefly. The relevant characteristics of the high temperature gas cooled reactor and fast breeder reactors are summarized. The use of 'off-peak' electricity to produce hydrogen by the electrolysis of water is considered. Various problems, political, social, organizational and technical are discussed.
- 4492 STREAMING PROBLEMS ARE THE MAJOR WORRY FOR SHIELDING EXPERTS. Martin, A. (Associated Nuclear Services, Epsom (UK)). Nucl. Eng. Int.; 22: No. 258, 30-31(Jun 1977).
The fifth international conference on reactor shielding was held in Knoxville, Tennessee, from 18 to 23 April, 1977 and was sponsored primarily by Oak Ridge National Laboratory. The design problems associated with LMFBR and fusion facilities were covered in many papers. Other aspects covered in detail were the recently encountered streaming problems and the increasing difficulties posed by activated corrosion products in LWR plants.
- 4493 TESTS IN SUPPORT OF LMFBR. Abrahamson, G.R.; Cagliostro, D.J.; Florence, A.L. (Stanford Research Inst., Menlo Park, Calif. (USA)). Nucl. Eng. Des.; 42: No. 1, 89-104(Jun 1977).
From International seminar on Containment of Fast Breeder Reactors; San Francisco, USA (22 - 23 Aug 1977).
Tests in support of LMFBR projects for potential incidents involving a hypothetical core disruptive accident or sodium-water interactions in steam generators have two possible goals: (1) to evaluate the integrity of the primary containment after the design is frozen or (2) to investigate design options in support of primary containment design. The test planning approach differs depending on the goal. The Fast Flux Facility tests had the first goal, the current Clinch River Breeder Reactor tests have the second goal. This paper discusses test planning, sources for simulating loads, modeling, and instrumentation. The main source described involves controlled venting of explosive gases. The source avoids generation of undesirable shock waves and facilitates calibration because the reaction rate of the explosive is independent of the response. It is indicated that the test program requiring the least time and cost involves a mixture of three types of models: small simple models, small complex models, and large complex models. Instrumentation is discussed that is required for validation of loading, response measurements for comparison with calculations, and response measurements on critical members where predictions are not possible.
- 4494 EXPERIMENTAL STUDY ON DYNAMIC RESPONSE OF REACTOR VESSEL TO AN HCDA. Ando, Y.; Kondo, S. (Tokyo Univ. (Japan). Faculty of Engineering); Yoshizawa, H. (Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan). Research and Development Center); Sasanuma, K. (Advanced Reactor Engineering Department, Atomic Power Division, Toshiba, Minatoku, Tokyo, Japan); Kawaguchi, O.; Mochio, T. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Nucl. Eng. Des.; 42: No. 1, 105-113(Jun 1977).
From International seminar on Containment of Fast Breeder Reactors; San Francisco, USA (22 - 23 Aug 1977).
This paper presents a discussion on the model experiments results for reactor structure dynamic response on FBR hypothetical core disruptive accident (HCDA) and the results of analysis using the dynamic response analysis code under experimental conditions. The purpose of this study is to clarify experimentally the dynamic response by use of scale models, as well as to attempt to confirm and improve the dynamic response analysis code on the basis of experimental data. The experimental results have clarified the inner barrel effects on reactor vessel deformation and its behavior due to a impact load. On the other hand, dynamic analysis was made of the 1/7.5 scale complex

model by a dynamic response analysis code 'PISCES-2DL', using the explosive combustion characteristics as inputs. Obtained values were compared with experimental values. Results showed that this method was fairly capable of evaluating radial deformation behavior in lower cylindrical parts of the vessel.

- 4495 CORE DEBRIS BEHAVIOUR AND INTERACTIONS WITH CONCRETE. Baker, L. Jr. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 42: No. 1, 137-150 (Jun 1977).

From International seminar on Containment of Fast Breeder Reactors; San Francisco, USA (22 - 23 Aug 1977).

The behavior of LMFBR core debris following a hypothetical core-disruptive accident (HCDA) depends upon a wide range of physical and chemical phenomena. Current understanding of the key phenomena are summarized for core-debris behavior within the reactor vessel, within the reactor cavity, and within the concrete base mat below the reactor cavity. In-vessel behavior was the principal consideration of post-accident heat removal for the FFTF reactor. Several concepts of engineered core-retention systems within the reactor cavity have been considered for other reactors, including the cooled crucible concept, the sacrificial barrier concept, the stable barrier concept, and the catch tray concept. Behavior within the concrete base mat is an important part of a general concept of inherent core retention which depends upon an understanding of the complex interactions of core debris with concrete.

- 4496 APPLICATION OF CONTAINMENT CODES TO LMFBRs IN THE UNITED STATES. Chang, Y.W. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 42: No. 1, 53-67 (Jun 1977).

From International seminar on Containment of Fast Breeder Reactors; San Francisco, USA (22 - 23 Aug 1977).

This paper describes the application of containment codes to predict the response of the fast reactor containment and the primary piping loops to HCDAs. Five sample problems are given to illustrate their applications. The first problem deals with the response of the primary containment to an HCDA. The second problem deals with the coolant flow in the reactor lower plenum. The third problem concerns sodium spillage and slug impact. The fourth problem deals with the response of a piping loop. The fifth problem analyzes the response of a reactor head closure. Application of codes in parametric studies and comparison of code predictions with experiments are also discussed.

- 4497 NUCLEAR ACCIDENT SIMULATION IN A 1/6 SCALE MODEL OF THE SNR-300 FAST BREEDER REACTOR. Egleme, M.; Fabry, J.P.; Lamotte, H. (Societe Belge pour l'Industrie Nucleaire, Brussels). Nucl. Eng. Des.; 42: No. 1, 115-122 (Jun 1977).

From International seminar on Containment of Fast Breeder Reactors; San Francisco, USA (22 - 23 Aug 1977).

In the frame of the licensing of the SNR-300 fast breeder reactor an HCDA has been investigated both by theoretical and experimental means. The latest phase of this programme was the simulation of a HCDA in a very well instrumented 1/6 scale model of the SNR-300 vessel. The Bethe-Tait accident was simulated by a slow burning charge. In order to check the reliability of the theoretical means allowing for an evaluation of these accidents, the German authorities wanted to receive,

before the shot, a theoretical precalculation report showing the main permanent deformations and responses of the pressure and strain gauges versus time. This paper compares and comments on the most meaningful results of this experiment with the SURBOUM-II precalculated estimations.

- 4498 ASSESSMENT OF ACCIDENT ENERGETICS IN LMFBR CORE-DISRUPTIVE ACCIDENTS. Fauske, H.K. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 42: No. 1, 19-29 (Jun 1977).

From International seminar on Containment of Fast Breeder Reactors; San Francisco, USA (22 - 23 Aug 1977).

An assessment of accident energetics in LMFBR core-disruptive accidents is given with emphasis on the generic issues of energetic recriticality and energetic fuel-coolant interaction events. Application of a few general behavior principles to the oxide-fueled system suggest that such events are highly unlikely following a postulated core meltdown event.

- 4499 TRANSIENT CONTAINMENT RESPONSE AND INHERENT RETENTION CAPABILITY. Glueckler, E.L.; Dayan, A.; Hayes, F.; Kline, C.T. (General Electric Co., Sunnyvale, Calif. (USA)). Fast Breeder Reactor Dept.). Nucl. Eng. Des.; 42: No. 1, 151-167 (Jun 1977).

From International seminar on Containment of Fast Breeder Reactors; San Francisco, USA (22 - 23 Aug 1977).

For safety assessments of early sodium cooled fast breeder reactors, hypothetical events such as loss of flow or transient overpower sequences combined with the failure of the reactor protective systems have been evaluated. The intent of these evaluations was to demonstrate that even extremely unlikely combinations of failures which could lead to meltdown of a portion of the reactor core would not propagate into events harmful to the public. This paper investigates the consequences of a meltdown event. In general, reactor vessels and the primary sodium systems have an adequate inherent retention capability to retain dispersed or molten core debris and remove decay heat. For certain vessel designs, large core debris masses may penetrate through the vessel wall and drain into the reactor cell. Although significant uncertainties exist in the evaluations of debris retention and heat removal capabilities of the containment, there is indication that criteria for radiological limits can well be met. If the accident progression can be controlled in-vessel, consequences would be minimized.

- 4500 APPLICATION OF A COUPLED EULER-LAGRANGE COMPUTER PROGRAM TO THE STRUCTURAL RESPONSE OF AN LMFBR. Hancock, S.L. (Physics International Co., San Leandro, Calif. (USA)). Nucl. Eng. Des.; 42: No. 1, 69-74 (Jun 1977).

From International seminar on Containment of Fast Breeder Reactors; San Francisco, USA (22 - 23 Aug 1977).

This paper discusses the application of PISCES 2DELK, a coupled Euler-Lagrange computer code, to the response of an LMFBR to a hypothetical core-disruptive accident. The coupled Euler-Lagrange method allows one to use an Euler finite difference mesh to compute fluid flow and a Lagrange mesh to compute structural response. The finite difference equations are discussed, and a sample calculation is presented in which the sodium gas bubble expands around a movable internal structure.

4501 TESTING PHILOSOPHY AND SIMULATION

TECHNIQUES. Holtbecker, H. (Commission of the European Communities, Ispra (Italy). Joint Research Centre). Nucl. Eng. Des.; 42: No. 1, 75-87 (Jun 1977).

From International seminar on Containment of Fast Breeder Reactors; San Francisco, USA (22 - 23 Aug 1977).

This paper reviews past and present testing philosophies and simulation techniques in the field of structure loading and response studies. The main objective of experimental programmes in the past was to simulate a hypothetical energy release with explosives and to deduce the potential damage to a reactor from the measured damage to the model. This approach was continuously refined by improving the instrumentation of the models, by reproducing the structures as faithful as possible and by developing new explosive charges. This paper presents an analysis of the factors which are expected to have an influence on the validity of the results e.g. strain rate effects and the use of water instead of sodium. More recently the discussion of a whole series of accidents in the probabilistic accident analysis and the intention to compare different reactor designs has revealed the need to develop and validate computer codes. Consequently experimental programmes have been started in which the primary aim is not to test a specific reactor but to validate codes. This paper shows the principal aspects of this approach and discusses first results.

4502 EXPERIMENTS ON SODIUM FIRES AND THEIR

AEROSOLS. Morewitz, H.A.; Johnson, R.P.; Nelson, C.T. (Atomics International Division, Rockwell International, Canoga Park, California, USA). Nucl. Eng. Des.; 42: No. 1, 123-135 (Jun 1977).

From International seminar on Containment of Fast Breeder Reactors; San Francisco, USA (22 - 23 Aug 1977).

A number of new sodium fire and aerosol experiments were undertaken to provide data for LMFBR safety analyses: (1) Experiments on the burning of single drops of liquid sodium falling in air have been performed to aid in model development for sodium spray fire codes. (2) The leakage of sodium oxide aerosols through a straight smooth capillary tube, representative of the maximum size of a hypothetical gas leak in the wall of the secondary containment of an LMFBR, has been studied. Even in those cases in which the capillary did not plug, <11% of the entering mass was of a respirable size as it emerged from the capillary. In addition, there were a number of conditions under which the capillary plugged. (3) Experiments on the behavior of high temperature, high concentration aerosols have shown a rapid depletion of the aerosol concentration in the first 6 sec following injection of approximately 800 g/m³ aerosols at approximately 1000°C into a closed vessel. This depletion has been correlated with the early formation of 100 to 200 µm agglomerates which fall out promptly.

4503 RESPONSE OF LIQUID METAL FAST BREEDER

REACTOR CONTAINMENT TO A HYPOTHETICAL CORE MELTDOWN ACCIDENT. Peak, R.D.; Simpson, D.E.; Stepnewski, D.D. (Hanford Engineering Development Lab., Richland, Wash. (USA)). Nucl. Eng. Des.; 42: No. 1, 169-180 (Jun 1977).

From International seminar on Containment of Fast Breeder Reactors; San Francisco, USA (22 - 23 Aug 1977).

This paper presents an analysis of the response of the containment building of a 2500-

MW liquid metal fast breeder reactor to a hypothetical reactor core meltdown. Although not mechanistically justifiable, this type of event is chosen for analysis as a basis for risk assessments. Containment space atmosphere compositions, temperatures, pressures, and structural temperatures are calculated, based on decay energy release and chemical reactions associated with the incident. The CACECO containment analysis code, which was used to make the calculations, is described in detail. Results of the study show that by utilizing the passive heat absorption capability of structures normally present in containment design, reactor plant containment integrity can be maintained for more than a day, even for extreme hypothetical events.

4504 FATIGUE CRACK PROPAGATION IN TYPES 304 AND 308 STAINLESS STEEL AT ELEVATED

TEMPERATURES. Raske, D.F.; Cheng, C.F. (Argonne Natl Lab, Ill). Nucl. Technol.; 34: No. 1, 101-110 (Jun 1977).

The fatigue crack-growth behavior of Type 304 stainless-steel base metal and Type 308 stainless-steel weld metal at elevated temperature was investigated using axially loaded single-edge-notch specimens. The crack-growth rates were determined and are presented as a function of the stress-intensity factor range. Both the base- and weld-metal specimens were tested in the as-received (or as-welded) and thermally aged condition. The results indicate that the crack-growth rates in the weld metal are significantly lower than in the base metal. In addition, aging at 593°C for 1000 h improved the resistance to fatigue crack growth in both the base and weld metals. These results are relevant to austenitic stainless steels fabricated by welding for use in liquid-metal fast breeder reactors.

4505 ROLE OF RELIABILITY AND RISK ASSESSMENT

IN LMFBR DESIGN: IMPLEMENTATION OF RELIABILITY IN LMFBR DESIGN. Graham, J.; Zemanick, P.P. (Westinghouse Electric Corp., Madison, Pa. (USA). Advanced Reactors Div.). Nucl. Eng. Des.; 42: No. 1, 1-9 (Jun 1977).

From International seminar on Containment of Fast Breeder Reactors; San Francisco, USA (22 - 23 Aug 1977).

This paper concerns future developments in LMFBR licensing technology. Federal Regulations (10 CFR 50.34) require that the preliminary safety analysis provide analysis and evaluation 'with the objective of assessing the risk to public health and safety' to determine margins of safety and the adequacy of the plant. Hitherto the assessment of risk has been qualitative but it has become increasingly apparent that quantitative assessments would provide a better basis for judgement. Potential future roles of reliability and risk assessment are discussed in the context of providing additional confirmation of the safety of LMFBR designs. Potential acceptability criteria for risk evaluations are outlined. The reliability implications of designing components to the ASME Code Section III requirements are discussed. General judgements are provided as well as the preliminary results of probabilistic studies of selected specific limits. There is a different reliability significance for the mandatory rules for normal, upset, and emergency conditions versus the non-mandatory rules for normal, upset, and emergency conditions versus the non-mandatory guidance for faulted conditions.

4506 RESULTS OF THE POSTIRRADIATION EXAMINATIONS OF THE RAPSODIE-1 EXPERIMENT FUEL

PINS. Haas, D. (Belgonucleaire, Brussels); van de Velde, J.; Gaube, M.; Ketels, J.; van Loon, C. Nucl. Technol.; 34: No. 1, 75-88 (Jun 1977).

An experiment sponsored by the Gesellschaft für Kernforschung at Karlsruhe, by Belgonucleaire, and by Interatom was carried out in the French reactor Rapsodie from February 1971 to September 1972, in the frame of a research and development program for the fast breeder prototype reactor SNR. This experiment consisted in the irradiation of two bundles, each containing 34 mixed-oxide pins, with spacer grids and tie-rods, up to a peak burnup of 10.6 at.% and a peak total neutron fluence of $8 \times 10^{20} \text{ mm}^{-2}$, corresponding to a peak fast fluence ($E > 0.111 \text{ MeV}$) of $6.4 \times 10^{20} \text{ mm}^{-2}$. The postirradiation examinations showed that the cold-worked WNr. 1.4970 cladding stainless steel has a lower irradiation-induced swelling than the solution-annealed WNr. 1.4988. Important inelastic deformations of both types of claddings have been observed; the largest part of these deformations can be explained by the in-pile creep. A mechanical interaction between the fuel and the cladding is not excluded in the case of the pins with the cladding steel WNr. 1.4970. The destructive examinations emphasized the influence of the pellets' initial stoichiometry on the fission product migration, on the plutonium radial distribution, and on the fuel-cladding chemical interaction. An interaction between the fuel and the blanket pellets, due to the cesium migration and to the formation of cesium uranates, leading to local cladding strains, has been noted. These phenomena, as well as the chemical interaction between the fuel and the cladding, have been investigated by extensive microprobe analysis. Finally, the fuel restructuring has been analyzed and interpreted as a function of the fuel characteristics, irradiation conditions, and cladding deformations.

- 4507 INFLUENCE OF THE RADIAL TEMPERATURE GRADIENT ON CLADDING DILATATION. Venker, H. (Institut fuer Material- und Festkoerperforschung, Karlsruhe, Ger.); Bober, M.; Schumacher, G. Nucl. Technol.; 34: No. 1, 98-100 (Jun 1977).

In the steep radial temperature gradient existing in the cladding wall of an LMFBR fuel pin, migration of vacancies up and lattice atoms down the temperature gradient can take place. The transport mechanisms are thermal diffusion and diffusion due to a radial activity gradient of vacancies caused by the temperature-dependent supersaturation of vacancies. An estimation shows that thermal diffusion could contribute to cladding dilatation under fast-neutron irradiation. The enhancement of diffusion under irradiation is derived from in-pile diffusion data of copper and gold in aluminum, because corresponding data of stainless steel are not available.

- 4508 HIGH-PRESSURE CAPSULE FOR THE IRRADIATION OF FUEL SWELLING SPECIMENS IN THE FR2 REACTOR. Haefner, H.E. (Inst. of Materials and Solid State Research, Karlsruhe, Ger.). Nucl. Technol.; 34: No. 1, 69-74 (Jun 1977).

Designing advanced fuel element concepts for fast breeder reactors and assessing by models the fuel rod behavior over the intended in-pile time require information about creeping and swelling in the nuclear fuels under in-pile conditions. A new series of experiments serves the concrete purpose of a quantitative assessment of the parameters of carbides as potential fast breeder fuels, carbide being

more susceptible to swelling than oxides. The loading of the specimen is given by the cladding restraint with the swelling fuel, and does not cause any undue expansion of the cladding of realistic fuel rods under operating conditions. This permissible contact pressure is generally below 300 bar (30 MPa). In an in-pile capsule recently developed, fuel specimens having the form of thin annular pellets may be exposed to omnilateral loads of up to 50 N/mm² by an aerostatic gas pressure of 500 bar (50 MPa) maximum at temperatures of up to approximately 1800°C. The swelling of the fuel after various burnups under different test conditions will be measured in a postirradiation examination. Because of the extraordinary stress exerted upon these high-pressure capsules under irradiation conditions, extensive preliminary tests and trials were necessary. A first prototype rig with two swelling specimen capsules containing five annular UO₂ pellets each has now been successfully tested in the FR2 reactor over an in-pile period of 170 days. In this case, the fuel temperatures were approximately 1350°C, and the specimen stresses were 45 N/mm². Further experiments are planned with (U, Pu)C-fuel up to 1800°C and 50 N/mm².

- 4509 PERSPECTIVES OF NUCLEAR ENERGY DEVELOPMENT IN EUROPEAN COUNTRIES. Zaleski, C.P.L. Adrikhalut; 37: No. 6, 36-39 (Jun 1977).

In 1975, nuclear energy represented almost 2% of energy production in E.E.C. countries. Projections for 1985 indicate 17% of the total energy production furnished by nuclear plants. Within a reasonable range of hypotheses, it seems that one should expect for the European countries a contribution of nuclear energy at the beginning of the next century at a level of between 25% and 50% of total energy consumption. This kind of development raises a serious problem of uranium resources, and shows the need, and even the urgency, of breeder reactors development. The various strategies of reactor development and its connections with uranium availability and cost are discussed. 8 refs.

- 4510 ACOUSTIC EMISSION IN SUBCOOLED NUCLEATE POOL BOILING. Nishihara, H.; Bessho, Y. (Kyoto Univ, Jpn). J. Nucl. Sci. Technol. (Tokyo); 14: No. 6, 407-415 (Jun 1977).

The acoustic emission in boiling had frequency components up to and sometimes beyond 50 kHz; the shape of power spectral densities behaved differently for heat fluxes above and below a certain value at which the overall acoustic intensity assumed a maximum value. With the increase in heat flux the acoustic emission increased in high frequencies below this heat flux, while it became eminent in low frequencies above the same heat flux. This phenomenon is related to the transition from the region with single bubbles to the region with coalesced bubbles. 23 refs.

- 4511 (JAPFNR--263) TEMPERATURE, FLOW, AND ACOUSTIC NOISES IN A LOCALLY BLOCKED 37-PIN BUNDLE. Kikuchi, Y.; Ogino, T.; Ozaki, Y.; Inushima, H. (Power Reactor and Nuclear Fuel Development Corp., Oarai (Japan); Mitsubishi Electric Corp., Amagasaki, Hyogo (Japan)). Jun 1977. Translation of PNC-N--943-77-05 (CONF-770640--2). 9p. Dep. NTIS (US Sales Only), PC A02/MF A01.

From 7. liquid metal boiling working group meeting; Petten, Netherlands (1 Jun 1977).

Work performed under United States--Japanese Fast Reactor Exchange Program.

Temperature, flow, and acoustic noises were measured in a locally blocked 37-pin bundle. Comparison of sodium temperature noise obtained in the blocked and unblocked experiments led to the observation that the blockage formation increased the measured root-mean-square (RMS) value by approximately three times in the low frequency ranges (0.1 approximately 10 Hz). Noncondensable gas (argon) entrainment caused a considerable increase in the outlet flow noise measured by an eddy-current type flowmeter. The boiling caused a considerable increase in acoustic noise intensity at all frequencies, particularly in the high frequency ranges (5 approximately 100 kHz).

- 4512 EVALUATION OF EFFECTS OF FINITE MIXING TIME AND NON-CONDENSABLE GAS ON FUEL--SODIUM INTERACTIONS. Saito, S. (Jpn At Energy Res Inst, Tokai-mura, Ibaraki-ken). J. Nucl. Sci. Technol. (Tokyo); 14: No. 6, 395-406(Jun 1977).

An approach has been tried to provide more realistic estimates of pressure history and mechanical work caused by fuel-sodium interaction (FSI) in postulated fast reactor accident. The work reported here is focussed on the evaluation of the effects due to fragmentation process or finite mixing time and presence of non-condensable gas on FSI. 6 refs.

- 4513 (JAPFNR--262) TRANSIENT BOILING OF SODIUM IN A 19-PIN BUNDLE UNDER LOSS-OF-FLOW CONDITIONS. Kikuchi, Y.; Haga, K. (Power Reactor and Nuclear Fuel Development Corp., Oarai (Japan)). Jun 1977. Translation of PNC-N--943-77-04 (CONF-770640--3). 9p. Dep. NTIS (US Sales Only), PC A02/MF A01.

From 7. liquid metal boiling working group meeting; Petten, Netherlands (1 Jun 1977).

Work performed under United States--Japanese Fast Reactor Exchange Program.

Experimental studies were carried out on transient sodium boiling in an electrically heated 19-pin bundle under loss-of-flow conditions. In the first series of experiments temperature distributions in the bundle were measured under initial steady state non-boiling conditions. The measured temperature distributions agreed fairly well with the calculation by the computer code NORMAL. In the second series of experiments transient boiling phenomena were investigated under loss-of-flow conditions. No high superheat was observed for boiling initiation. The observed coolant voiding was initially limited to the center subchannel because of steep temperature gradient in the bundle and then spread slowly.

- 4514 DETERMINATION OF THE BENDING LOADINGS RESULTING FROM THE SUPERHEATED STEAM FLOW PATH VARIATION IN AN EXPANSION PIPE-BEND FOR SODIUM-WATER HEAT EXCHANGERS. Crucitti, N. (Ansaldo S.p.A., Genoa (Italy)). Energ. Nucl. (Milan); 24: No. 6, 327-334(Jun 1977).

The structural design of an asymmetric stiffened pipe incorporating an expansion bend, as applied in a sodium-water heat exchanger, was presented in *Energia Nucleare*, vol. 23(10) p. 507 (October 1976). The present article describes a further theoretical approach required to make such design more complete and reliable. In particular, the bending loadings exerted on the pipe-bend, as resulting from the path variation of the running fluid, are evaluated. All correlations associated to such loadings are derived in terms of the expansion bend sizing parameters and superheated steam flow properties and presented in a quick-use form for practical design purposes.

- 4515 EVALUATION OF EFFECTS OF FINITE MIXING TIME AND NON-CONDENSABLE GAS ON FUEL SODIUM INTERACTIONS. Saito, S. (Japan Atomic Energy Research Inst., Tokai, Ibaraki. Tokai Research Establishment). J. Nucl. Sci. Technol. (Tokyo); 14: No. 6, 395-406(Jun 1977).

An approach has been tried to provide more realistic estimates of pressure history and mechanical work caused by fuel-sodium interaction (FSI) in postulated fast reactor accident. The work reported here is specifically focussed on the evaluation of the effects due to fragmentation process or finite mixing time and presence of non-condensable gas on FSI. As one of the methods to describe the fragmentation process in FSI, the mass of fuel which interacts with sodium has been introduced as a function of time. The presence of non-condensable gas in the unheated sodium regions above and below the interaction region as well as in the interaction region itself has been taken into account in the model, which affects value of sonic velocity in the former, and volume change in the latter. An extensive parametric study has led the conclusion that the total mechanical work as well as a value of peak pressure at the initial period in which sodium remains liquid strongly decrease with the mixing time constants and/or non-condensable gas content in the system, and in the most plausible case, a possible maximum value of peak pressure through the entire period in the fuel-sodium interaction appears to be at most a few hundred atmospheres.

- 4516 DEVELOPMENT OF FAST REACTORS AND STUDIES ON MATERIALS. Mishima, Y. (Tokyo Univ. (Japan). Faculty of Engineering). Kinzoku; 47: No. 6, 7-12(Jun 1977). (In Japanese).

The effort of independent development has born fruit, and the fast experimental reactor "Joyo" has attained the criticality on April 24, 1977. The fast neutron source reactor in the University of Tokyo has been in operation since 1972, but Joyo is the first full-scale fast reactor. The refining techniques for sodium and the purity control for sodium in use were important problems. The cooling of fuel rods, the swelling of fuel due to high burn-up, the strict dimensional allowance for fuel assemblies owing to poor controllability, the supporting method without large flow resistance for fuel assemblies, the local overheat of claddings due to FP gas leak, and the irradiation damage of claddings were deemed to be the requirements. The method of manufacturing fuel pellets was investigated. The temperature of coolant has been set at 500 deg. C reluctantly because of the difficulties in the development of cladding material. Steam-cooled fast reactors were conceived to avoid delay, but were not materialized. The cladding material for Joyo is 316 stainless steel containing reduced nitrogen as the measures to helium damage. As the piping material, 2,1/4 Cr-1Mo steel without niobium was used. In Japan, nickelclad vanadium alloy has been investigated in the University of Tokyo to give sodium resistance with nickel cladding.

- 4517 FLOW IN SODIUM LOOP SURGE TANK. Matál, O.; Martoch, J. (Vyzkumny Ústav Zavodu Energetického Strojirenství, Brno (Czechoslovakia). Odbor Nukleární Techniky). Jad. Energ.; 23: No. 6, 207-211(Jun 1977). (In Czech).

The alternate liquid flow, the condition of vortex formation, gas entrainment in the discharge and the liquid level characteristics are studied using the models of the vertical and horizontal surge tanks of a sodium circuit

with pump and heat exchangers. The conditions for vortex formation are more favourable in the vertical cylindrical tank than in the horizontal tank. The size of the vortex produced in the tank is affected by the initial speed circulation, due as a rule to an unsuitable inlet design. The proposed design considers an inlet below the sodium level using capped perforated pipes. Vortex formation, gas transport to the discharge pipe and turbulences of the liquid in the tank may be prevented by dividing the tank to the discharge and the inlet areas using perforated partitions, and by inserting the discharge cylinder above the discharge pipe outflow. The liquid level in the tank may be damped by screens or by perforated plates. The adaptation of the surge tank of the sodium circuit will probably eliminate vortex formation and the entrainment of cover gas into the discharge piping and the sodium circuit under nominal conditions.

and accumulate the international competitive power in the economy of future practical reactors. The general construction and the reactor core of this loop type 300 MWe plant are described. SUS 316 stainless steel is used for the handling head, upper shield, cladding tubes, entrance nozzle, and wrapper tube, and molybdenum-clad tantalum is considered as the lower shield in core fuel assemblies. In blanket fuel assemblies, the diameter of cladding tubes is different, and the lower shield is made of SUS 316. SUS 304 stainless steel is used for neutron shields, surveillance assemblies, and neutron sources. However, the wrapper tubes for the surveillance assemblies and neutron sources are made of SUS 316. SUS 316 or 304 is used for the most parts of control assemblies except the controlling material of B₁C. Trial manufacture, irradiation test and other tests are carried out to develop the materials for Monju.

- 4518 FEW-GROUP CONSTANTS FOR CALCULATING $K_{SUB}(EFF)$ AND $\Delta(1/K_{SUB}(EFF))$ OF FAST REACTORS. Svarny, J. (Skoda, Pízen (Czechoslovakia)). *Zavod Vystavba Jadernych Elektraren*. *Jad. Energ.*; 23: No. 6, 201-207 (Jun 1977). (In Czech).

Theoretical and numerical analyses are made of the behaviour of the linear and bilinear averaging of group constants aimed at error accumulation in few-group reactivity coefficient ($k_{sub}(eff)$) calculations and of their first order perturbations ($\Delta(1/k_{sub}(eff))$) as a result of inaccuracies in weight functions. The results were confirmed using the BN-600 fast reactor model. The obtained results are in full agreement with theoretically derived relations.

- 4519 NATURAL EVAPORATION OF SODIUM UNDER THE CONDITIONS OF MIST FORMATION. Kumada, T.; Ishiguro, R. (Hokkaido Univ., Sapporo (Japan)). *Faculty of Engineering*; Abe, T. *Nippon Kikai Gakkai Ronbunshu*; 43: No. 370, 2278-2287 (Jun 1977). (In Japanese).

The natural evaporation of liquid sodium from a horizontal isothermal plane is described. Experiments were carried out by changing the temperature of liquid sodium or the pressure of argon gas. Estimation of the amount of evaporation was also made by the critical supersaturation model (CSM). The experimental results were compared with the theoretical estimation. The rate of evaporation was 1 to 3 times as large as that estimated from the analogy of heat transfer and material transfer, and was almost same as the result of forced evaporation experiment. The estimated values of vapor pressure by the CSM were accurate enough for practical use. However, this agreement does not mean the adequacy of assumption of the CSM. For the further discussion, more information concerning the mist formation is required.

- 4520 MATERIALS FOR FAST BREEDER PROTOTYPE REACTOR 'MONJU'. Kitagawa, H. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). *Kinzoku*; 47: No. 6, 18-22 (Jun 1977). (In Japanese).

The Power Reactor and Nuclear Fuel Development Corp. is forwarding the design of the fast breeder prototype reactor 'Monju'. The purposes of the development of Monju are to obtain the prospect on the performance, safety, reliability and operation of FBR power plants by the independent development and through the experiences of design, manufacture, construction and operation, in order to put FBRs in practical use in 1990s, and to develop

- 4521 BEHAVIOR OF CHROMEL-ALUMEL THERMOCOUPLES DURING EMERGENCY SHUTDOWN OF THE BR-10 REACTOR. Kruglov, A.S.; Vyrodov, P.V.; Redchenko, M.I. *At. Energ. (USSR)*; 42: No. 6, 504-505 (Jun 1977). (In Russian).

ALUMEL; CHROMEL; DATA; HIGH TEMPERATURE; PHYSICAL RADIATION EFFECTS; SBR-5 REACTOR; SCRAM; THERMOCOUPLES; THERMOELECTRICITY; TIME DEPENDENCE; PERFORMANCE

- 4522 BEHAVIOUR OF CHROMEL-ALUMEL THERMOCOUPLES DURING EMERGENCY SHUTDOWN OF THE BR-10 REACTOR. Kruglov, A.S.; Vyrodov, P.V.; Redchenko, M.I. *At. Energ. (USSR)*; 42: No. 6, 504-505 (Jun 1977). (In Russian).

ALUMEL; CHROMEL; DATA; HIGH TEMPERATURE; PHYSICAL RADIATION EFFECTS; SBR-5 REACTOR; SCRAM; THERMOCOUPLES; THERMOELECTRICITY; TIME DEPENDENCE

- 4523 LIQUID-METAL DIP SEAL WITH PNEUMATIC SPRING. Poindexter, A.M. (to Energy Research and Development Administration). *US Patent* 4,026,763. 31 May 1977. Filed date 12 Apr 1976. 4p.

PAT-APPL-675,845.

An improved liquid-metal dip seal for sealing the annulus between rotating plugs in the reactor vessel head of a liquid-metal fast-breeder nuclear reactor has two legs of differing widths communicating under a seal blade; the wide leg is also in communication with cover gas of the reactor and the narrow leg is also in communication with an isolated plug annulus above the seal. The annulus contains inert gas which acts as a pneumatic spring. Upon increasing cover gas pressure which depresses the level in the wide leg and greatly increases the level in the narrow leg, the pneumatic spring is compressed, and resists further level changes, thus preventing radioactive cover gas from bubbling through the seal.

- 4524 SUPERPHENIX. A BREEDER REACTOR IN FULL SCALE. Lingjaerde, R. (Institutt for Atomenergi, Kjeller (Norway)); Vendryes, G.A. *Tek. Ukebl.*; 124: No. 22, 12, 14, 16 (20 May 1977). (In Norwegian).

A description of the Super Phenix reactor is preceded by a brief discussion of the background for the development of the breeder reactor and of the basic principles of the fast breeder. In conclusion the time schedule for Super Phenix is presented and the future prospects for fast breeder nuclear power plants in France are briefly discussed.

- 4525 TERTIARY HEAT EXCHANGER. Hahn, G. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). German (FRG) Patent 2,550,329/A/. 18 May 1977. 9p. (In German).

The invention deals with the improvement of a tertiary heat exchanger possessing a contact medium besides a heating and working medium. These kind of heat exchangers are used in sodium-cooled nuclear power plants. The contact medium used so far should be replaced by a metal granulate which is used in porous bulk of single grains of varying size. It is advantageous to use copper as granulate.

- 4526 REACTOR ROTATING SHIELD WITH CONVECTION BARRIER FOR METAL VAPOURS. Rothfuss, H.; Koehler, M. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). German (FRG) Patent 1,813,820/C/. 18 May 1977. 2p. (In German).

This construction for liquid metal cooled reactors is to prevent the ingress of sodium vapour, for example, into the gap between the rotating shield and the reactor tank, without making this gap extremely small or constantly heating it. This is achieved by separating the functions of rotation and sealing. For sealing, the rotating shield when stationary, i.e. in normal operating conditions, is set on a flanged ring welded inside the tank, so that sodium vapour cannot pass upwards through this tank zone and gradually block up the space above. For rotation, i.e. during fuelling and fuel unloading periods of the reactor, the rotating shield is lifted by several lifting cylinders from this sealing surface. These lifting cylinders sit on a geared ring surrounding the top of the rotating shield, which is supported on the circumference on roller bearings and is sealed sideways against the surrounding tank wall by moving seals. The requirements for these seals are less than those for the stationary seal described above, as sodium temperatures and evaporation are appreciably reduced during refuelling periods compared with normal operation.

- 4527 FAST BREEDER REACTOR. Ziegler, A. (to Siemens A.G., Berlin (Germany, F.R.); Siemens A.G., Muenchen (Germany, F.R.)). German (FRG) Patent 1,948,912/C/. 5 May 1977. 6p. (In German).

The invention deals with a fast breeder reactor with liquid metal cooling and with fuel zone co-axially surrounded by a breeder zone without a separating wall, where the coolant is admitted to the whole cross-section of the breeder reactor with the same pressure. It is difficult in such nuclear reactors to maintain a constant pressure uniformly over the entire reactor cross-section. According to the invention, this is achieved in that one arranges the fuel elements as well as the breeder elements in open construction and designs flow-conducting devices arranged on top of one another which influence the coolant flow. The design of this flow-conducting device is part of the patent claims.

- 4528 IMPROVED NUCLEAR FUEL ASSEMBLY GRID SPACER. Marshall, J.; Kaplan, S. (to Energy Research and Development Administration). US Patent 4,021,300. 3 May 1977. Filed date 23 May 1975. 10p.

PAT-APPL-580,498.

Disclosed is an improved fuel assembly grid spacer and method of retaining the basic fuel rod support elements in position within the fuel assembly containment channel. The

improvement involves attachment of the grids to the hexagonal channel and forming the basic fuel rod support element into a grid structure, which provides a design which is insensitive to potential channel distortion (ballooning) at high fluence levels. In addition the improved method eliminates problems associated with component fabrication and assembly.

- 4529 DRYOUT HEAT FLUXES FOR INDUCTIVELY HEATED PARTICULATE BEDS. Dhir, V.; Catton, I. (Univ. of California, Los Angeles). J. Heat Transfer; 99: No. 2, 250-256 (May 1977). From Winter annual meeting of the American Society of Mechanical Engineers; Houston, TX, USA (30 Nov 1975).

Experimental observations of the dryout heat fluxes for inductively heated particulate beds have been made. The data were obtained when steel and lead particles in the size distribution 295 to 787 microns were placed in a 4.7-cm dia pyrex glass jar and inductively heated by passing radio frequency current through a 13.3-cm dia multiturn work coil encircling the jar. Distilled water, methanol and acetone were used as coolants in the experiments, while the bed height was varied from 1.9 to 8.9 cm. Different mechanisms for the dryout in deep and shallow beds have been identified. Dryout in shallow beds is believed to occur when the vapor velocity in the gas jets exceeds a certain critical velocity at which choking of the vapor, leading to obstruction in the flow of the liquid towards the bed occurs. However, deep beds dry out when gravitational force can no longer maintain a downward coolant flow rate necessary to dissipate the heat generated in the bed. Finally, the heat flux data of the present investigation and that from two previous investigations made at Argonne Laboratory and at UCLA have been correlated with semitheoretical correlations based on the proposed hydrodynamic models.

- 4530 STUDY OF POST-ACCIDENT MOLTEN FUEL DOWNWARD STREAMING THROUGH THE AXIAL SHIELD STRUCTURE IN THE LIQUID-METAL FAST BREEDER REACTOR. Gasser, R.D.; Kazimi, M.S. (Brookhaven National Lab., Upton, NY). Nucl. Technol.; 33: No. 3, 248-259 (May 1977).

A two-dimensional transient freezing model is developed to study the dynamics of solidification of a fluid flowing in a cylindrical channel, the walls of which are cooled below the freezing temperature of the fluid. The model is applied to the case of molten fuel flowing downward through the coolant channels in the lower shielding structure of current liquid-metal fast breeder reactor designs subsequent to a hypothetical core meltdown accident. The results indicate that under post-accident conditions, a high potential exists for rapid relocation of significant quantities of core debris across the shield structure.

- 4531 NONEQUILIBRIUM EVAPORATION AND CONDENSATION IN LIQUID-METAL FAST BREEDER REACTOR FUEL EXPANSION. Refling, J.G.; Reynolds, A.B.; Garner, P.L.; Rao, S.P. (Univ. of Virginia, Charlottesville). Nucl. Technol.; 33: No. 3, 275-289 (May 1977).

The effect of nonequilibrium mass transfer on two-phase oxide fuel expansion was studied. An analytical model was applied to fuel expansion tests being planned at Oak Ridge National Laboratory (ORNL) and to whole-core expansion of a uniform bubble in Clinch River Breeder Reactor geometry. Among the parameters varied in the study are the liquid droplet

radius ($R/\text{sub } p$) and the condensation and evaporation coefficients (σ). It was found that allowance for nonequilibrium mass transfer increased the expansion times for the ORNL tests when the ratio of $\sigma/R/\text{sub } p$ was $<1000 \text{ m}^{-1}$. Nonequilibrium mass transfer had little effect on the whole-core expansions until $\sigma/R/\text{sub } p$ reached unrealistically low values of $<10 \text{ m}^{-1}$.

4532 CARTER PUTS NUCLEAR POWER IN STRONG POSITION. Smock, R.W. Electr. Light Power (Boston); 55: No. 5, 11-12(May 1977).

As far as nuclear power is concerned, the most positive factor of the National Energy Plan is support for light-water reactors, coupled with the limitations imposed on fossil-fuel generation. On the negative side, the ban on commercial nuclear fuel reprocessing and the halt on construction of the Clinch River Breeder Reactor appear to limit the long-term prospects for nuclear power. The best news the author says, is Carter's awareness that energy supply, despite conservation, must grow. He also announced plans for a new uranium enrichment plant, badly needed to support LWR growth. Other portions of the Carter program ban natural gas as a utility fuel, raise the price of oil, and boost the environmental control costs of burning coal. If economic studies based on pre-policy conditions favored nuclear generation over coal or oil, they will overwhelmingly favor nuclear now. The administration made a number of vague proposals for the nuclear industry--that licensing be streamlined and safety standards be tightened, for example--but it remains to be seen how they will be implemented. On balance, the new Carter program gives nuclear a green light, Mr. Smock says. Some opinions extracted from papers presented at the American Power Conference on the President's policy regarding the breeder reactor are reviewed. One of the most hotly debated issues at the conference dealt with proliferation, with almost all speakers on the subject saying Carter's policy will not work. (MCW)

4533 CARTER PLAN GENERALLY BACKED EXCEPT ON BREEDERS, SCRUBBERS. Electr. Light Power (Boston); 55: No. 5, 1-2(May 1977).

Electric utilities see President Carter's energy plan as a yellow light to move cautiously the way they've been going: to more nuclear and coal-fired generation, to load management via new rate structures, to promotion of insulation programs, and toward incorporation of new energy sources and technologies into traditional utility systems. But the utilities see scrambled signals in Carter's plans to increase coal and nuclear power because they don't like requiring all new coal units to have scrubbers and they find it ironic that the President's conservation-oriented approach sacrifices the energy potential of plutonium. It is the general thrust of the program, the appeal to conservation, that is most important, along with the turn away from dependence on oil and natural gas for most of our energy needs. Specifics of the package will be in flux until Congress adopts it.

4534 IMPACT OF THE NEW ENERGY TECHNOLOGIES. Decker, G.L. (Dow Chem Co, Midland, Mich). Mech. Eng.; 99: No. 5, 24-27(May 1977).

According to a Ford Foundation report that "assumes that the nation will not deliberately impose any policies that might affect our ingrained habits of energy use, but will make a strong effort to develop supplies at a rapid

pace to match rising demand", U.S. energy demand in the year 2000 would reach 185 quads (a quad is defined in this article as one quadrillion Btu= 10^{15}) if historical trends were to continue. That is almost 2 1/2 times present use. Probably none of the so-called "wonder solutions"--solar energy, geothermal, burning wastes, coal gasification, etc.--will be ready for any meaningful contribution by that date. Only by intense exploitation of just about every known energy source (with special emphasis on coal), in addition to conservation of energy in every possible way, can economic stagnation and considerable change for the worse in the American life style be avoided. The article includes a chart forecasting the expected contribution in quads of energy, and the approximate date of (U.S.) commercialization of new energy technologies, for the period 1975 to 2000. A second chart compares the proportional contributions of 1972 technology and year 2000 technologies, including conservation and higher efficiencies. 6 refs.

4535 WORK STARTS IN SUPERPHENIX AT THE CREYS-MALVILLE SITE. Banal, M. Nucl. Eng. Int.; 22: No. 257, 41-45(May 1977).

Details are given of the 1200 MWe SuperPhenix fast breeder reactor. Design principles are outlined and essential differences between SuperPhenix and Phenix are discussed: the main differences are the new heat exchanger design and the containment system. Descriptions are given of the site layout, core, nuclear steam supply system, main secondary circuits and the fuel handling system. Safety features which are discussed include licensing procedure, core surveillance, decay heat removal and the containment system. A tables of parameters is included.

4536 STRUCTURAL ANALYSIS TECHNOLOGY FOR HIGH-TEMPERATURE DESIGN. Greenstreet, W.L. (Oak Ridge National Lab., Tenn. (USA)). Nucl. Eng. Des.; 24: No. 3, 375-386(May 1977).

Results from an ongoing program devoted to the development of verified high-temperature structural design technology applicable to nuclear reactor systems are described. The major aspects addressed by the program are (1) deformation behavior; (2) failure associated with creep rupture, brittle fracture, fatigue, creep-fatigue interactions, and crack propagation; and (3) the establishment of appropriate design criteria. This paper discusses information developed in the deformation behavior category. The material considered is type 304 stainless steel, and the temperatures range to 1100°F (593°C). In essence, the paper considers the ingredients necessary for predicting relatively high-temperature inelastic deformation behavior of engineering structures under time-varying temperature and load conditions and gives some examples. These examples illustrate the utility and acceptability of the computational methods identified and developed for predicting essential features of complex inelastic behaviors. Conditions and responses that can be encountered under nuclear reactor service conditions and invoked in the examples.

4537 RECENT DEVELOPMENTS IN EQUIPMENT FOR SODIUM TEST LOOPS AND FAST REACTORS. Orszak, P. Nucl. Eng. Int.; 22: No. 257, 53-55(May 1977).

Since completion of PFR, work of GEC Reactor Equipment Ltd. on sodium has developed further and a standard range of sodium components is being marketed. These include

level monitoring equipment, flow-meters, a wide range of electro-magnetic pumps, automatic plugging meters, cleanup units, filters, pressure vessels and valves. In order to test and prove these products the Company has established its own sodium test facility. Special test loops have been constructed to establish the performance, durability and reliability of components and for equipment calibration.

- 4538 CAREFUL ATTENTION TO DETAIL WAS NECESSARY IN DEVELOPING THE SUPERPHENIX STEAM GENERATORS. Robin, M.G. Nucl. Eng. Int.; 22: No. 257, 46-48(May 1977).

The development, design, and testing of the SuperPhenix steam generator are described. The generator is a once-through unit with economizer, evaporator, and superheater located along the same tube. Its features are illustrated, including the method of tube fastening and spacing. The material for the generator is 316 L stainless steel, and the tube bundle material Alloy 800. Development tests were conducted to investigate thermal and flow performance, tube penetrations under thermal shock, fatigue and wear of expansion bends, sodium flow and tube vibration, and safety in case of steam tube failure.

- 4539 OPERATION OF PROTOTYPE FAST REACTOR STEAM GENERATORS LED DIRECTLY TO COMMERCIAL-SIZE DESIGN. Taylor, D. Nucl. Eng. Int.; 22: No. 257, 49-53(May 1977).

The design and manufacture of the PFR steam generators are outlined. Experiences with leaks in evaporators and superheaters during operation are described. These have led to certain design changes for the steam generators for the commercial demonstration fast reactor, which are discussed. The main changes are: adoption of ferritic steel in place of austenitic to avoid stress corrosion and caustic cracking after a leak; elimination of welded joints to simplify manufacture and minimize the chance of leaks; redesign of the header system to eliminate thick tube plates and thermal shock problems. These changes can be accommodated in the 'U' tube type of generator but in place of the forced circulation system adopted for PFR a 'U' tube partial recirculation boiler with separate evaporator and superheater is proposed for the commercial fast reactor.

- 4540 INSTRUMENTATION AND CONTROL SYSTEMS ON PFR. McAffer, N.T.C. J. Inst. Nucl. Eng.; 18: No. 3, 66-73(May-Jun 1977).

An account is given of the instrumentation and control systems of the Prototype Fast Reactor at the UKAEA Dounreay Reactor Establishment under the following headings - outline of the plant; general requirements; main aspects of the overall system; some particular aspects (data reduction equipment, the automatic protective system, sodium measurements, hydrogen detection in the secondary sodium circuit, temperature measurements, examples of interlock and control logic, cabling and power supplies); and operating experience and possible future developments.

- 4541 STRESS ANALYSIS OF A NUCLEAR REACTOR GRID PLENUM ASSEMBLY AND ITS APPLICATION TO THE EBR-II. Chang, L.K.; Marchertas, A.H. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 24: No. 3, 387-397(May 1977).

A structural model is presented to predict the stress and deformation of a nuclear reactor

grid plenum assembly. The model consists of two circular grid plates on top of one another which are interconnected by a large number of tubes, a perforated cylindrical shell and a cylinder at the outer edge of the plates. During normal reactor operation and shutdown the grid plenum assembly is subjected to both external and internal loads, in which the external load is due to the dead weight of the subassemblies resting on the upper grid plate and the coolant pressure, while the internal load is due to the temperature gradient and irradiation swelling in the grid plates accumulated through the life of the reactor. The structure is divided into many annular regions implying an axisymmetric treatment. The solution of the structural system is obtained by solving the equilibrium equations of the individual tubes, plates and shells, and satisfying the compatibility condition at the boundary of each region. The swelling strain in the grid plates is treated analogously to that of thermal strain in the elastic region. The numerical results of the problem are obtained by means of a computer code which uses a matrix-inversion technique to solve the simultaneous algebraic equations involved.

- 4542 SAFETY OF THE FAST REACTOR. Matthews, R.R. (Central Electricity Generating Board, London (UK). Nuclear Health and Safety Dept.). Atom (London); No. 247, 86-89(May 1977).

From Meeting at Imperial Coll. of Science and Technology; London, UK (15 Jan 1977). Verbatim of an address by R.R. Matthews, Chief Nuclear Health and Safety Officer, UK Central Electricity Generating Board given on January 15th 1977. The object of this address was to give some opinions on the safety issues of fast reactors as seen from an operational point of view. An outline of the basic responsibilities for nuclear safety is first given, and it is emphasized that the Central Electricity Generating Board has a statutory responsibility for the safe operation of its nuclear plant. The Nuclear Installations Act places absolute responsibility on the operator for ensuring that injury to persons and damage to property do not occur, and the new Health and Safety at Work Act does likewise. In addition the Board has a Nuclear Health and Safety Department that has to ensure that adequate provision for safety is made in the design, construction, and operation of nuclear plant, and safety at operational stations is monitored continuously by inspectors. In addition the requirements of the Nuclear Installations Inspectorate, laid down in the site licence conditions, must be satisfied. All these requirements are here discussed in the light of application to commercial fast reactors. It is considered that the hazards to fast reactor operating personnel are small and little different from those of other types of reactor, and in some respects the fast reactor has advantages, particularly in regard to the use of a Na coolant. The possibility of various types of accident is considered. Radioactive effluent discharge is also considered. The fast reactor as an international problem is discussed, including security matters. The extensive experience gained in operation of the experimental and prototype fast reactors at Dounreay is emphasized.

- 4543 ANALYSIS OF EBR-II SECONDARY SODIUM NETWORK FOR PRESSURE PULSES DUE TO LEAKS OF STEAM ON WATER INTO SODIUM. Srinivas, S.; Chopra, P.S. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 24: No. 3, 313-325(May 1977).

A systematic procedure is presented for the safety analysis of the secondary sodium system of an LMFBR in the event of a leak of water or steam into sodium in an evaporator or a superheater. Using fracture mechanics, it is shown that the usual assumption of failure initiation by guillotine rupture of one or more water or steam tubes is unrealistic. A model is proposed for the gradual growth of leaks due to the phenomenon of wastage. The pressure rise in the system due to a sodium-water reaction is calculated solving one-dimensional hydrodynamic equations. By comparing results obtained for the proposed gradual growth model with those due to guillotine failure it is shown that the assumption of guillotine failure leads to a significant overestimation of pressures and stresses. Based upon the proposed leak progression model the stresses in the EBR-II secondary sodium system are shown to be within safe limits.

- 4544 HYPOTHETICAL ACCIDENT RULES FBR SAFETY. REPORT ON THE FAST BREEDER REACTOR SAFETY CONFERENCE IN CHICAGO. Pepler, W.; Royl, P. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Reaktorentwicklung). Atomwirtschaft, Atomtech.; 22: No. 5, 270(May 1977). (In German).
From International meeting on fast reactor safety and related physics; Chicago, Ill. (5 - 8 Oct 1976).
LMFBR TYPE REACTORS; INFORMATION; REACTOR SAFETY

- 4545 METHOD FOR VAPORIZING MIXED OXIDE FAST REACTOR FUELS FOR ANIMAL INHALATION STUDIES. Allen, M.D.; Kaune, W.Y.; Craig, D.K. (Battelle Pacific Northwest Labs., Richland, Wash. (USA)). Health Phys.; 32: No. 5, 389-396(May 1977).

High-powered lasers were used to vaporize zirconium dioxide and uranium dioxide to show the feasibility of laser vaporization of UO_2 - PuO_2 fast reactor fuels to generate condensation aerosols for animal inhalation studies. Three types of pulsed lasers were used: a CO_2 gas laser with an energy output of 0.2 J/pulse, a ruby laser with an output of 8 to 10 J/pulse, and a neodymium-YAG laser with an output of approx. 35 J/pulse. An incandescent vapor plume resulted from each pulse for all three types of lasers, with the plume size dependent on beam energy. The aerosol, which was visually observed for each type of laser, was collected on membrane filter paper. The amount of aerosol collected on the filter paper per pulse was directly proportional to the energy density of the beam. Scanning electron micrographs of the aerosol showed branched chains composed of spherical particles of 0.1 μm diameter which are typical of the vaporization-condensation aerosols of most solids. X-ray diffraction analyses showed that the aerosol was primarily amorphous, with diffraction patterns from a few of the samples indicating simple cubic structure. Lack of structure in most of the samples was attributed to the fast quench of the laser plume.

- 4546 PHYSICS AND ENGINEERING CHARACTERISTICS OF A LMFBR IN LONG-TERM OPERATION. Kujal, J.; Jakab, J. (Ustav Jaderneho Vyzkumu, Rez (Czechoslovakia)). Jad. Energ.; 23: No. 5, 162-168(May 1977). (In Czech).

The results are presented of calculating the main physics and engineering parameters of a 1600 MWe fast breeder reactor. The changes in the parameters during long-term operation at a selected realistic fuel cycle are studied. Homogeneous isotope concentrations in the FBR

reactor core and breeding blankets in the initial state, and homogeneous concentrations of heavy fuel isotopes in all reactor regions before and after the 6th refuelling are tabulated. The mean burnup of fuel removed from the core under steady state conditions is 60 MWd/kg U+Pu, 3.5 MWd/kg U+Pu for the front breeding blanket and 5.6 MWd/kg U+Pu for the side breeding blanket. The course of breeding ratio is graphically shown for the entire reactor and partial breeding ratios are listed for the core, front and side breeding blankets. Changes were studied in the power distribution along the reactor radius and its levelling up during long-term operation is shown. The mathematical model of the reactor is briefly described as are the basic calculation procedures. Actual operating conditions of the reactor are considered, such as fractional discontinuous refuelling and reactor operation at a given output with $k_{sub}(eff)=1$.

- 4547 THERMAL OPENING OF IRRADIATED URANIUM OXIDE FUEL ELEMENTS OF BR-5 REACTOR WITH FUEL SEPARATION BY SOLUTION. Novoselov, G.P.; Bibikov, S.E. At. Energ. (USSR); 42: No. 5, 398-399(May 1977). (In Russian).
Deposited article; for English translation see the journal Sov. At. Energy.
DECLADDING; DISSOLUTION; FLUE GAS; FUEL CANS; NITRIC ACID; SBR-5 REACTOR; SPENT FUEL ELEMENTS; URANIUM DIOXIDE; ZONE REFINING

- 4548 DETECTION OF LIQUID METAL COOLANT BOILING-UP. Aleksandrov, K.A.; Afanas'ev, V.A.; Gataullin, N.G.; Golushko, V.V. At. Energ. (USSR); 42: No. 5, 410-411(May 1977). (In Russian).
ACOUSTIC EMISSION TESTING; BACKGROUND NOISE; BOILING DETECTION; BOR-60 REACTOR; REACTOR COOLING SYSTEMS; HIGH TEMPERATURE; KHZ RANGE 01-100; KHZ RANGE 100-1000; PIEZOELECTRICITY; SODIUM; SPECTRA; TRANSDUCERS

- 4549 IRRADIATION BEHAVIOUR OF CARBIDE FUEL ELEMENTS IN THE BOR-60 BREEDER REACTOR. Tsykanov, V.A.; Gryazev, V.M.; Davydov, E.F.; Kuz'min, V.I. (and others). At. Energ. (USSR); 42: No. 5, 378-382(May 1977). (In Russian).
The results of an investigation of irradiation behaviour of uranium carbide fuel elements in the BOR-60 reactor up to burnups of 10 per cent fima are presented. The fuel elements and fuel assemblies characteristics, as well as the irradiation history are shortly described. The most stringent irradiation mode was as follows: neutron fluence of 6×10^{22} neutr/cm² ($E > 0.1$ MeV), linear power density of 700 W/cm, fuel cladding temperature of 680 deg C. The axial core distribution as well as dependency of geometrical parameters of the stainless steel fuel cladding on fuel burnup and density were measured. To attain zero plastic deformation of the fuel cladding, total volume of voids in the fuel, and a radial gap between the fuel pellets and the cladding is indicated to be 20 per cent. The average rates of fuel swelling at fuel element core temperatures about 1200 ± 50 deg C were measured to be equal to 1.5 ± 0.2 per 1 per cent of burnup with respect to density variations. Good compatibility between the fuel and the cladding is underlined.

- 4550 TEST RESULT FOR THE CONTROL ROD DRIVE MECHANISM IN SODIUM FOR PROTOTYPE LMFBR 'MONJU'. Ishii, Y.; Murakami, Y. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Doryokuro Giho; No. 22, 69-84(May 1977). (In Japanese).

The Japanese prototype LMFBR (liquid metal fast breeder reactor) "Monju" has nine rough control rods, three fine control rods, four safety rods and three back-up stopping rods, nineteen rods in total. These control rods must have very high reliability for the function of regulating reactivity, scramming, etc.. The Power Reactor and Nuclear Fuel Development Corporation ordered in 1971 the manufacture of three control rod drive mechanisms, the design of which was a little different each other, and these were tested in water first, then tested in sodium at the "Osai Engineering Center" one by one. About 500 times of scram test, and 10000 to 100000 times of driving test in full stroke and about 50 mm eccentric condition were carried out. These test results were very satisfactory, consequently second step of trial manufacture was conducted by reflecting the results of the tests and modifying the mock-ups. The arrangement of control rods in the core of "Monju", the functions of each rod, the design specification of the mock-ups of this mechanism, the structural design, the test facility in sodium, the test results, the characteristic evaluation and the feedback of these results to the design of the "Monju" mechanism are explained in this paper.

- 4551 ONE-DIMENSIONAL ANALYSES ON GENERATION AND PROPAGATION OF PRESSURE WAVES IN COMPRESSIBLE LIQUID RESULTING FROM FUEL COOLANT INTERACTION. Miyazaki, K.; Fujii, Y. (Osaka Univ., Suita (Japan). Faculty of Engineering). J. Nucl. Sci. Technol. (Tokyo); 14: No. 5, 327-342 (May 1977).

The pressure response obtained for a vertical column is identical to that resulting from the acoustic constraint model (which is adopted by Cho et al.) within the range of time required for the waves to traverse and retrace the column at $t < l/c$ (t is time, l is length of liquid column, c is propagation velocity of pressure wave), but beyond this time point, the present analysis yields the response that differs according to the history of the waves. On the other hand, for the spherical compression waves propagating symmetrically outwards in coolant, the solution of this pressure is obtained using the step function and $\exp(-ct/R)$ (R is radius of fuel coolant interaction boundary). It is suggested that the predicted pressure should be reduced appreciably by adopting the rigorous solution of the spherical acoustic constraint model, combined with realistic values chosen for the interaction time.

- 4552 CENTRAL COMPUTERIZED DATA BASE FOR LMFBR SAFETY CODES. Greene, N.M.; Flanagan, G.F.; Alter, H. (Oak Ridge National Lab., TN). CODATA Bull.; No. 23, 29-31 (May 1977).

From 5. international CODATA conference; Boulder, CO, USA (Jun 1976).

The status of the development of the SACRD (Safety Analysis Computerized Reactor Data) data base is described. The overall data base effort is concerned with gathering, evaluating, and distributing all necessary nondesign-related data required to perform safety analysis. These analyses cover a wide range of situations: initiating events and protected accidents, loss of subassembly integrity, core disassembly, structural response, aerosol transport and biological dose, and post-accident heat removal. The data in SACRD are structured by type, property, material, and version.

Shiraha, M.; Sakano, K.; Hayashi, S.; Hatakeyama, G. (Kawasaki Heavy Industries Ltd., Kobe (Japan)). FAPIG (Tokyo); No. 85, 15-21 (May 1977). (In Japanese).

The steam generator takes an important position specifically in the development of sodium-cooled fast breeder reactors. Though the electrically heated test rig of 0.1 MW for steam generators was already produced and tested, the new 0.1 MW sodium-heated test rig for straight tube type steam generators has been manufactured to forward the development of straight tube steam generators strongly. The testing purpose of the rig is to grasp the characteristics in the departure from nucleate boiling phenomenon caused on the water/vapor side under sodium-heating condition. At present, the static characteristic is being tested among the following planned tests: (1) static test (departure from nucleate boiling test, heat transfer test), (2) dynamic characteristic test, (3) flow stability test, (4) water leak detector test, (5) endurance test of sodium-immersed rupture disc, (6) thermal shock test of tube plates. The rig is composed of water and sodium systems of closed loops, and the straight tube type steam generator. Each part of the rig (steam generator, water system, sodium system, control system, alarm, interlock, cover gas) is equipped with a variety of detectors. From the results of trial run and static characteristic test, the time required to reach 100% load from start-up is about 15 hours, and 8 hours are required from 100% load to shut down. An example of the static characteristic showed that the gradient of sodium temperature distribution changed remarkably at the part 6.6 m downward from the sodium inlet. This shows that the departure from nucleate boiling occurred at the place. Calculation supported the above estimation.

- 4554 INFLUENCE OF COLD WORK ON THE IN-PILE STRESS RUPTURE STRENGTH OF 16 CHROMIUM--16 NICKEL--NIOBIUM AUSTENITIC STAINLESS STEELS. Closs, K.D.; Schaefer, L.

(Kernforschungszentrum, Karlsruhe, Ger.). J. Test. Eval.; 5: No. 3, 201-209 (May 1977).

From 8. international symposium on the effects of radiation on structural materials; St. Louis, MO, USA (4 May 1976).

The in-pile stress rupture strength of 16Cr--16Ni--Nb stainless steel was investigated. Different pretreatments were used, namely solution annealing plus various cold-working levels up to 20 percent. Thermal stability under irradiation and the high-temperature embrittlement of the different material conditions were studied. The specimens were irradiated in the form of electrically heated pressurized tubes at irradiation temperatures ranging from 615 to 720°C. Irradiation time was up to 4800 h, corresponding to a maximum fast neutron fluence ($E > 0.1$ MeV) of 8.2×10^{25} n/m². Neutron irradiation caused a marked reduction in stress rupture strength for the solution-annealed and the cold-worked specimens. For the solution-annealed tubes, this reduction was mainly attributed to high-temperature embrittlement; the decrease in rupture strain was less pronounced for the cold-worked tubes. The reduction in stress rupture strength came mainly from an increase in creep rate. As far as stress rupture strength and creep strength at high irradiation temperatures were concerned, the 15% cold-worked 16Cr--16Ni--Nb steel was superior to the other cold-work levels.

- 4553 TEST RIG OF STRAIGHT TUBE TYPE SODIUM-HEATED STEAM GENERATOR. Yoshizaki, A.;

4555 MATRIX HEAT EXCHANGER. Fewell, T.E.; Ward, C.T. (to Energy Research and

Development Administration, Washington, D.C. (USA)). German (FRG) Patent 2,647,653/A/. 28 Apr 1977. 17p. (In German).

Heat exchangers for nuclear reactors cooled with liquid metal are often of matrix construction. The invention deals with the improvement of the heat transfer behaviour of such aggregates. A small gap between the matrix and pipeline is to be filled with a liquid medium of good thermal conductivity (e.g. melted metal such as Bi, Pb, Hg or their alloys). Additions of Zr, Mg or Ti are to prevent the cooling pipe walls from corrosion. Advantageous variations of the composition and arrangement are given in detail.

4556 DEVICE TO DETECT GASES, IN PARTICULAR FISSION GASES, IN THE COOLING CHANNEL OF A LIQUID-COOLED NUCLEAR REACTOR FUEL ELEMENT. Gast, K. (to Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). German (FRG) Patent 1,807,801/C/. 28 Apr 1977. 2p. (In German).

In order to control sodium-cooled fast nuclear reactors, the early discovery and localization of cladding damage on individual fuel elements is important. It is thus suggested to create rotational movements in the cooling channel of a fuel element and to insert a heated thermoelement as detecting device in the cyclone centre. The thermoelement can be electrically heated or by embedment in a body containing fissionable material. The equipment invented is highly sensitive and has a short response time.

4557 PROCESS FOR COOLING A BREEDER REACTOR AND BREEDER REACTOR FOR CARRYING OUT THE PROCESS. Gast, K. (to Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). German (FRG) Patent 1,940,014/B/. 14 Apr 1977. 3p. (In German).

The invention concerns the guiding of the coolant flow in a sodium cooled breeder nuclear reactor. The coolant is first taken upwards from below through the breeding elements of the radial breeding zone, is diverted downwards to the inflow area of the fission zone and there flows through the fuel elements upwards from below in the well known way. A breeder reactor laid out for this flow is described.

4558 METHOD FOR CLEANING COMPONENTS CONTAMINATED PREFERABLY WITH RADIOACTIVE SODIUM AND FACILITY TO APPLY THE METHOD. Schultheiss, G.F.; Menzenhauer, P. (to Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)); Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,545,031/A/. 14 Apr 1977. 9p. (In German).

An improvement of the method for cleaning components of nuclear installations contaminated with radioactive sodium is proposed. The method refers to the way of concentrating the wash solution and is to cause the amounts of liquid used to be reduced. The facility is described in detail. There are used water and steam under protective gas.

4559 NUCLEAR REACTOR EQUIPPED WITH A CONTROL MECHANISM. Ingham, R.V. (to UKAEA London Office). French Patent 2,348,545/A/. 12 Apr 1977. Priority date 13 Apr 1976, United Kingdom of Great Britain and Northern Ireland (UK), 8p. (In French).

The invention relates to a control mechanism for an LMFR type reactor, to ensure a fast power drop by varying the reactivity of an appropriate amount to meet the conditions of

continuous operation in complete safety should there be a partial drop in the flow or pressure of the coolant. Control rods in a neutron absorbing material are used to regulate the reactivity of the reactor. At least one control rod is sensitive to a triggering signal so as to be introduced in the reactor core in two successive stages, first by uncontrolled unidirectional injection and, secondly, by controlled introduction. The uncontrolled unidirectional injection may be carried out by free drop or by propulsion under the effect of stored energy.

4560 PROCESS AND APPARATUS FOR DETECTING THE POSITION OF A FAILED TUBE IN A TUBULAR LIQUID METAL-WATER HEAT EXCHANGER. Schrock, S.L.; Mangus, J.D. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). French Patent 2,348,483/A/. 12 Apr 1977. Priority date 12 Apr 1976, United States of America (USA), 16p. (In French).

The invention concerns a process and apparatus for detecting the position of a leak in a tube of a tubular heat exchanger of the liquid metal-water type, habitually employed in a sodium cooled nuclear reactor. Under the invention an inert gas is let in on the outside of the tubes, a pressure difference is created along the tube under test, the lower pressure taking place inside the tubes. The tube is then heated to bring about the dissociation and isolation of the reaction products which are then analysed to indicate the faulty tubes.

4561 HEAT EXCHANGER FOR HIGH TEMPERATURES AND PRESSURES. (to CEA, 75 - Paris (France)). French Patent 2,386,798/A/. 5 Apr 1977. 15p. (In French).

This invention refers to a heat exchanger for high temperatures and pressures. Specifically, the invention concerns a high temperature and high pressure heat exchanger of the type comprising, within a leaktight containment, a plurality of tubes around which flows a primary fluid and through which flows a secondary fluid, and at least one plate delimitating an outlet chamber for the secondary fluid at one of the ends of the containment. This exchanger prevents the establishment of excessive thermal gradients in the wall of this chamber, when rapid changes in the temperature of this secondary fluid occur.

4562 EVALUATED DELAYED NEUTRON SPECTRA AND THEIR IMPORTANCE IN REACTOR CALCULATIONS. Saphier, D. (Argonne National Lab., IL); Ilberg, D.; Shalev, S.; Yiftah, S. Nucl. Sci. Eng.; 62: No. 4, 660-694 (Apr 1977).

Delayed neutron emission spectra from thermal-neutron fission of ^{233}U , ^{235}U , ^{239}Pu , and ^{241}Pu , from fast-neutron fission of ^{232}Th , ^{235}U , ^{238}U , and ^{239}Pu and from high-energy neutron (14.7-MeV) fission of ^{235}U and ^{238}U , for six groups of delayed neutrons are evaluated. The evaluation is based on recent measurements of delayed neutron spectra from 20 fission product isotopes. The data are presented in graphic form and are compared to directly measured equilibrium spectra whenever available. Tables with a convenient 54-energy-group structure are provided to facilitate their use in reactor calculations. The results of a limited number of two-dimensional, multigroup, transient calculations for the Clinch River Breeder Reactor core, using the newly evaluated spectra, are compared with calculations using some older spectra. The importance of the inclusion of these data in reactor dynamic calculations is evaluated. 34 figures, 13 tables. 37 references.

- 4563 COSTS AND FINANCING OF ADVANCED REACTOR TYPE DEVELOPMENT. EXCERPT. Technol.-Nachr. Manage.-Inf.; No. 158, 1-16(Apr 1977). (In German).

The Federal Ministry for Research and Technology (BMFT) presented on March, 16th, 1977 the development programs for power reactors, their costs and their financing to the Budget Committee of the Bundestag. By means of this presentation, the BMFT is facing the opinion, that the costs management of huge armament programs (MRCA) could be transferred to huge technological projects set up by the BMFT. The presentation of finance concerning the development of reactors is extracted from the BMFT's paper (accentuated by special type setting).

- 4564 SIMPLE-2: A COMPUTER CODE FOR CALCULATION OF STEADY-STATE THERMAL BEHAVIOR OF ROD BUNDLES WITH FLOW SWEEPING. Jones, O.C. Jr.; Yao, S.C.; Henry, R.E. (Argonne National Lab., Ill. (USA)). Nucl. Eng. Des.; 41: No. 2, 205-217(Apr 1977).

A computer code has been developed for use in making single-phase thermal hydraulic calculations in rod bundle arrays with flow sweeping due to spiral wraps as the predominant crossflow mixing effect. This code, called SIMPLE-2, makes the assumption that the axial pressure gradient is identical for each subchannel over a given axial increment, and is unique in that no empirical coefficients must be specified for its use. Results from this code have been favorably compared with experimental data for both uniform and highly nonuniform power distributions. Typical calculations for various bundle sizes applicable to the LMFBR program are also included.

- 4565 ELECTROMAGNETIC PUMP FOR FBR. Namba, M.; Mukaida, H.; Taguchi, J.; Miyahara, M.; Miyagi, T. (Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Toshiba Rebyu; 32: No. 4, 314-317(Apr 1977). (In Japanese).

The development of liquid metal fast breeder reactors and research about related matters are being pushed forward in many countries. As liquid metal sodium is generally used, and for its recirculation use is made of mechanical pumps when the flow rate is high, and electromagnetic pumps when the flow rate is low. Toshiba, manufacturing various types of the latter, has obtained rich data on actual operation with sodium test facilities. Herein outlined are flat linear induction type and the annular linear induction type of them.

- 4566 UTILIZATION FACTOR OF NUCLEAR MATERIAL IN LMFBR. Tomabechi, K. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). J. Nucl. Sci. Technol. (Tokyo); 14: No. 4, 314-316(Apr 1977).

BREEDING BLANKETS; BURNUP; EFFICIENCY; FABRICATION; FUEL CYCLE; FUEL MANAGEMENT; LMFBR TYPE REACTORS; NUCLEAR FUELS; NUCLEAR MATERIALS MANAGEMENT; PLUTONIUM; REPROCESSING; USES

- 4567 SODIUM VAPOR DEPOSITION ONTO A HORIZONTAL FLAT PLATE ABOVE LIQUID SODIUM SURFACE. II. ANALYSIS OF SODIUM VAPOR DEPOSITION RATE WITH SODIUM MIST GROWTH WITHIN MAIN FLOW REGION. Kudo, K. (Hitachi Ltd., Ibaraki (Japan)). Hitachi Research Lab.; Hirata, Masaru. Nippon Kikai Gakkai Ronbunshu; 43: No. 368, 1407-1417(Apr 1977). (In Japanese).

The sodium vapor deposition onto a horizontal flat plate above liquid sodium surface was studied. The analysis was performed by assuming that the sodium mist is emitted into the main flow without condensation and then grows up in the main flow and drops on the sodium surface. The effects of growth of sodium mist to the system were investigated. The model of the phenomena is the sodium deposition onto a horizontal flat plate which is placed above the sodium surface with the medium cover gas. One-dimensional analysis can be done. The rate of deposition is greatly reduced when the temperature of the flat plate is lowered. For the analysis of this phenomena, it is assumed that the sodium mist grows by condensation. One of results is that the real state may be the state between the state that the condensation of mist is made in the boundary layer and the state that the mist is condensed in the main flow. Others are that there is no effect of sodium mist condensation on the rate of deposition, and that the rate of the vaporization of sodium is given by the original and the modified model.

- 4568 APPROACH TO SELECTED STRUCTURAL MECHANICS PROBLEMS OF THE FAST POWER REACTOR CORE. Dienstbier, J.; Pazdera, F. (Ustav Jaderneho Vyzkumu, Rez (Czechoslovakia)). Jad. Energ.; 23: No. 4, 140-146(Apr 1977). (In Czech).

Thermal deformations, swelling and thermal creep result in dimensional changes and deformations of fuel assemblies in the core and the side breeding blanket. The method of calculating bending deformations of the fuel assemblies is based on the theory of slim statically indefinite beams calculated using the matrix technique. The beam is divided into elements and linear temperature distribution and neutron flux distribution along the element length are assumed. The calculation method was used in compiling programme PA 2147 in the GIER ALGOL language, simulating straining and deformations of the individual assemblies during reactor operation. The programme also covers the calculation of assembly cross section increase due to thermal dilatation, swelling and to elastic creep deformation by sodium overpressure in the assembly. The results are briefly described of the behaviour of the BN-600 fast reactor assemblies and of the strength deformation calculations of assemblies of a 1600 MWe fast power reactor. A toroidal model is shown for the description of strength and deformation conditions in the spacing level of fast reactor assemblies. The model was used in writing programme PA 2132 in the GIER ALGOL language. The results show that stress and deformation conditions in the fast sodium cooled power reactor core approach limit conditions, which is mainly due to the swelling of structural material, i.e., austenitic steel.

- 4569 RESEARCH AND DEVELOPMENT ON STEAM GENERATOR FOR FAST BREEDER REACTOR. Tanaka, K. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Karyoku Genshiryoku Hatsuden; 28: No. 4, 321-335(Apr 1977). (In Japanese).

The sodium-heated steam generator for a liquid metal fast breeder reactor is one of the most important components, but many problems exist, such as the possibility of sodium-water reaction in the steam generator, the heat transfer characteristics of sodium, the flow instability of water inside heat transfer tubes, the choice of material, etc. Much activity is concentrated for the research and development of the steam generator for "Monju" plant which is the first prototype

fast breeder, electricity-generating plant in Japan. The basic flow chart, the layout of the plant, the key parameters of the steam generator for 'Monju' plant are explained. For steam generators, many concepts for heat transfer tubes have been devised, for example helical coil tube, straight tube, U tube, hair pin tube, etc., and each type has advantages and also disadvantages. The research and development works for the steam generator are reviewed in general, and 50 MW steam generator test and many sodium-water reaction tests are described in detail, which have been carried out at Oarai Engineering Center of the Power Reactor and Nuclear Fuel Development Corporation.

- 4570 WASTAGE PHENOMENA DUE TO SMALL LEAK SODIUM-WATER REACTION. REACTION TEMPERATURE AND WASTAGE PATTERN. Nei, H. (Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)); Hori, M. Nippon Kikai Gakkai Ronbunshu; 43: No. 368, 1418-1425 (Apr 1977). (In Japanese).

A theory was developed for analyzing the wastage phenomena due to the sodium-water reaction. The theory is based on the viewpoint that the wastage is primarily due to corrosion. Equations for temperature and concentration distributions of reaction product jet were derived the first place and secondly, an equation for the temperature dependence of corrosion was derived. The results of theoretical analysis based on these equations were compared with the results of wastage experiments performed by the present authors and that performed at APDA (Atomic Power Development Associates). The results of authors' experiments are described together with the experimental conditions and used apparatus. Nozzle diameter, sodium-water temperature, and the water pressure were varied as parameters. As for theoretical analysis, descriptions are given of the analytical model, basic equations for jet flow accompanying chemical reactions, the shape of surface reaction, and the temperature distribution of jet flow. Finally, the mechanism of wastage is considered and discussed based on the theoretical analysis and comparison with the results of experiments.

- 4571 MECHANICAL PROPERTIES OF LIQUID-METAL FAST BREEDER REACTOR PRIMARY PIPING MATERIALS. Brinkman, C.R.; Sikka, V.K.; King, R.T. (Oak Ridge National Lab., TN). Nucl. Technol.; 33: No. 1, 76-95 (Apr 1977).

The available tensile and creep-rupture properties of relevant product forms of Types 316 and 304 stainless steel and associated as-deposited weld metals (i.e., Type 316) are reviewed. A description of both time-dependent and independent behavior is emphasized for Type 316 stainless steel, since this material and its weldments in hot-leg piping applications must operate at temperatures within the creep range. Therefore, elevated-temperature fatigue and creep-fatigue interaction of Type 316 stainless steel are discussed over the temperature range from 566 to 593°C for both air and sodium environments.

- 4572 EFFECT OF CHEMISTRY VARIATIONS ON THE SHORT-TERM RUPTURE LIFE AND TENSILE PROPERTIES OF 20% COLD-WORKED TYPE 316 STAINLESS STEEL. Duncan, D.R.; Paxton, M.M. (Hanford Engineering Development Lab., WA). Nucl. Technol.; 33: No. 1, 112-119 (Apr 1977).

The effects of compositional variations on the rupture life of 20% cold-worked Type 316 stainless steel were investigated at 19-ksi (131-MPa) uniaxial tensile stress and at 1400°F

(1033 K). Forty-nine different alloys were studied, with compositional variations from nominal in carbon, nitrogen, phosphorus, sulfur, boron, manganese, copper, silicon, molybdenum, cobalt, chromium and nickel. This alloy and cold-work level represents the duct and fuel cladding material choice for the first four core loadings of the Fast Flux Test Facility, a key element in the Liquid-Metal Fast Breeder Reactor Program. Tensile properties of four of the alloys were studied at temperatures from room temperature to 1600°F (1144 K). Boron, nitrogen, and molybdenum plus silicon additions significantly increased rupture life, while chromium and carbon additions decreased rupture life. Molybdenum plus silicon additions increased yield and ultimate strength and ductility at 1200°F (922 K) and below.

- 4573 (DOE-tr--177) LASER-DOPPLER MEASUREMENTS OF ENERGY DENSITY SPECTRA IN TURBULENT CHANNEL FLOW. Loercher, G. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). Inst. fuer Reaktorentwicklung). Apr 1977. Translation of KFK--2448 122p. Dep. NTIS, PC A06/MF A01. Work performed under United States--W. Germany Fast Reactor Exchange Program.

The results are presented of measurements in a turbulent channel flow. The measurements are carried out with a laser-Doppler anemometer. The fluid is water. Both the main flow components of the velocity and a component perpendicular to it are measured. The parameters are the Reynolds number and the distance from the wall. Especially the energy density spectra of the velocity fluctuations are evaluated. These spectra are considered in detail in this investigation. The spectra are subdivided into three ranges. The range of lower frequencies or wave numbers is represented by an approximate equation, and the parameters in the equation are calculated. The -1 pattern in this range is also evaluated.

- 4574 MAGNETOSTRICTIVE PROBE. Knaak, J. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). German (FRG) Patent 2,541,530/A/. 31 Mar 1977. 11p. (In German).

The probe consists of a rod-shaped magnetostrictive material with two coils axially wound around it. One coil serves as a measuring coil whenever pressure is applied axially on the rod, the other coil is for premagnetization of the rod. The front end of the rod is connected via bellows to a diaphragm, this constitutes the front part of the cylindrical probe-housing. The back end of the rod is connected to the housing by a sound-absorbing material. The device is used as a sound sensor, for example in the liquid sodium cooling circuit of a nuclear reactor.

- 4575 PROCEDURE AND CIRCUIT ARRANGEMENT FOR LEAK DETECTION. Fuge, R.; Pridoehl, E.; d'Avignon, H.; Froehlich, K.J.; Uhlmann, G. German (GDR) Patent 130,289/A/. 25 Mar 1977. 8p. (In German).

Procedure and circuit arrangement for detection of small leaks in sodium heated steam generators of fast breeder reactors are described. The noise arising from the liquid sodium-water reaction is received by microphones and the acoustic signals are processed with the aid of the following devices: amplifier, filter, variable-gain amplifier, discriminator, counter, comparator, and processor. In case of exceeding a certain threshold value an alarm system is switched on.

4576 EQUIPMENT FOR DETECTING FISSION

PRODUCTS. Jacobi, S. (to Deutsches Patentamt, Muenchen (Germany, F.R.); Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). German(FRG) Patent 2,037,053/C/. 24 Mar 1977. 4p. (In German).

A carrier gas charged with fission products is taken into a cylindrical precipitation vessel with a bottle-neck opening. There is an electromagnet on the bottle-neck which deflects the ionised solid fission products caesium and rubidium to a detector. The β and/or γ radiation emitted by the precipitated particles is measured with a detector. The wall of the precipitation vessel consists of non-ferromagnetic material in the area of the electromagnet. The measurement is still possible in the presence of metal vapour (sodium as coolant) and/or the presence of water vapour.

4577 REVERSE STEAM GENERATOR. Dubsek, F.; Srutek, J.; Sychrova, H.; Masek, V. Czech Patent 168,265/B/. 15 Mar 1977. 5p. (In Czech).

A steam generator is proposed for a liquid metal-cooled fast reactor power plant only featuring the primary circuit filled with liquid metal and lacking the liquid metal-filled secondary circuit. For such design, a so-called reverse steam generator should be used in which a failure of one heat transfer tube does not cause failures of other tubes. On the opposite, the intensity of the accident decreases with time or the liquid metal reaction with water or steam eventually stops. The proposed reverse steam generator design eliminates the disadvantages of the existing types resulting from very different thermal expansion values for the individual steam generator parts. For this purpose, a steam separation equipment is placed in the evaporating zone body of the steam generator, tubes are spaced in a square lattice and are shaped into bends forming an expansion compensator.

4578 ARRANGEMENT FOR LEAK DETECTION. Fuge, R.; Pridoehl, E.; d'Avignon, H.; Froehlich, K.J.; Uhlmann, G. German(GDR) Patent 130,080/A/. 11 Mar 1977. 5p. (In German).

An arrangement for the detection of microleaks in sodium heated steam generators of fast breeder reactors is described. A vessel with a free sodium surface is installed in the sodium circuit. In case of a microleak water and sodium react to form hydrogen, which escapes in the form of bubbles within the vessel. The noise resulting therefrom is received by one or several microphones located inside or outside the vessel.

4579 EQUIPMENT FOR TESTING A GROUP OF NUCLEAR

REACTOR FUEL ELEMENTS FOR DAMAGE TO THE CANS. Moha, F. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). German(FRG) Patent 2,537,393/A/. 3 Mar 1977. 15p. (In German).

Equipment is described for use in sodium cooled nuclear reactors, with which the fuel elements consisting of bundles of fuel and fertile rods can be examined for damage to the cans. Fission products occurring in the liquid coolant act as indicators. The coolant is sucked via pipelines which penetrate into the elements into a collecting container, and a special pipeline is available for every element

of a group, where the highest points of individual pipelines at different hydrostatic heads are taken to the collecting container. This permits the checking of one line at a time due to pressure changes.

4580 FUEL ELEMENT CLUSTER OF A NUCLEAR REACTOR. Kurilkin, V.V.; Kusmartsev, E.V. German(FRG) Patent 2,443,018/B/. 3 Mar 1977. 6p. (In German).

The suggestion deals with spacers in fuel elements for fuel element clusters used in fast nuclear reactors with liquid metal cooling. In order to counteract possible deformation at high power such as swelling etc., it is suggested to strengthen the wire wound around the jacket surface of each fuel element as spacer, with 2 wires each so that a wire bundle is formed which is rigidly joined on the sections between the contact planes.

4581 (JAPFNR--260) PROGRESS REPORT ON FAST BREEDER REACTOR DEVELOPMENT IN JAPAN, OCTOBER--DECEMBER 1976. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Mar 1977. Translation of PNC-N--251-77-09 10p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

Included are brief sections dealing with Joyo construction, Monju design, fast reactor kinetics, reactor components, reactor instrumentation, sodium technology, fuel development, reactor safety, and steam generator development. (DG)

4582 IN-REACTOR STRESS RELAXATION IN BENDING OF 20% COLD-WORKED 316 STAINLESS STEEL. Kenfield, T.A.; Busboom, H.J.; Appleby, W.K. (General Electric Co., Sunnyvale, Calif. (USA). Breeder Reactor Dept.). J. Nucl. Mater.; 65: No. 1, 238-243(Mar 1977).

From International conference on measurement of irradiation-enhanced creep in nuclear materials; Petten, The Netherlands (5 - 6 May 1976).

A test is in progress to measure in-reactor stress relaxation of 20% cold-worked 316 stainless steel in bending through the sequential irradiation in the experimental reactor EBR-II of two materials capsules. This paper details the results from the first capsule, the second capsule has completed irradiation and is awaiting examination. Approximately 50% total relaxation was measured following irradiation to $2 \times 10^{21} \text{ n/cm}^2$, $E > 0.1 \text{ MeV}$ at 370°C . The extent of relaxation was independent of specimen orientation to rolling direction, independent of initial stress level and only weakly dependent on neutron dose at this fluence level.

4583 BUILDING STOP FOR KALKAR FAST BREEDER REJECTED. Energiewirtsch. Tagesfragen; 27: No. 3, 234-236(Mar 1977). (In German).

The Administrative Court at Duesseldorf, with its decision of January 20th, 1977 - 3 L 1337/76 - has rejected the application of a high-school teacher in Moers for 'restitution of the claim's suspensive effect'. The costs of the procedure were awarded against the applicant. The decision also confirmed the exclusion of belated objections. The main reasons upon which the judgment is based are presented in full.

4584 (KFK-tr--522) FISSION PRODUCT MONITORING IN THE SODIUM COOLANT DURING OPERATION OF A FAST REACTOR. Ivanov, V.B.;

Polyakov, V.I.; Chechetkin, Yu.V.; Shipilov, V.I. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Internationales Buero/Uebersetzergruppe). Mar 1977. Translation of Operativnyi kontrol produktov deleniya v natrievom teplositele bystrogo reaktora 5p. (In German). Dep. NTIS (US Sales Only), PC A02/MF A01.

A Compton integral-gamma spectrometer is described serving to monitor the state of the core of fast reactors and to assess the activity of the coolant during operation.

- 4585 OPTIMIZATION OF REFUELING SCHEME OF FAST REACTOR CORE BY MEANS OF MAN MACHINE INTERACTIVE METHOD. Kobayashi, Y.; Kondo, S.; Togo, Y. (Tokyo Univ. (Japan). Faculty of Engineering). J. Nucl. Sci. Technol. (Tokyo); 14: No. 3, 177-194 (Mar 1977).

A computer system for performing quick survey and optimization has been developed to aid in-core fuel management of sodium cooled fast reactors. The method utilizes the conversational mode of computer operation, to perform on-line computation, display of results and acceptance of commands, thus permitting rapid exchange of information between the machine and the person in charge. To improve the overall efficiency of the system as problem-solver, an algorithm is introduced to provide for automated shuffling to determine the fuel loading patterns of successive cycles. This algorithm serves to model the refueling and shuffling arrangement after a prescribed optimized standard core pattern according to the extent of burnup of each fuel sub-assembly. The proposed system is applied to problems of: (1) straightforward simulation of fuel performance in given in-core fuel management programs, covering core and radial blanket refueling and control rod positioning, (2) optimization, and (3) modification of in-core fuel management programs. Several numerical examples are treated, to confirm the applicability of this conversational-mode problem-solving system to in-core fuel management.

- 4586 REACTOR VESSEL STRUCTURAL RESPONSE TESTS AT FBR HYPOTHETICAL ACCIDENT. Ando, Y.; Kondo, S. (Tokyo Univ. (Japan). Faculty of Engineering); Kawaguchi, O.; Kano, T.; Takei, A. Nippon Genshiryoku Gakkaishi; 19: No. 3, 177-184 (Mar 1977). (In Japanese).

Dynamic response characteristics of liquid metal cooled fast breeder reactor vessel structure subjected to hypothetical core disruptive accident load was investigated by a series of 1/15 and 1/7.5 scaled model tests using a specifically developed chemical explosive for pressure source simulating the postulated load condition. Dynamic pressure and strains were measured successfully at several points of the models and a lot of interesting and useful results were obtained under realistic loading conditions. General characteristics of the structural response was clearly shown by the test results.

- 4587 SMALL LEAK DETECTION BY MEASURING SURFACE OSCILLATION DURING SODIUM-WATER REACTION IN STEAM GENERATOR. Nei, H.; Hori, M. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). J. Nucl. Sci. Technol. (Tokyo); 14: No. 3, 200-209 (Mar 1977).

Small leak sodium-water reaction tests were conducted to develop various kinds of leak detectors for the sodium-heated steam generator in FBR. The super-heated steam was injected into sodium in a reaction vessel having a

sodium free surface, simulating the steam generator. The level gauge in the reaction vessel generated the most reliable signal among detectors, as long as the leak rates were relatively high. The level gauge signal was estimated to be the sodium surface oscillation caused by hydrogen bubbles produced in sodium-water reaction. Experimental correlation was derived, predicting the amplitude as a function of leak rate, hydrogen dissolution ratio, bubble rise velocity and other parameters concerned, assuming that the surface oscillation is in proportion to the gas hold-up. The noise amplitude under normal operation without water leak was increased with sodium flow rate and found to be well correlated with Froude number. These two correlations predict that a water leak in a 'MONJU' class (300 MWe) steam generator could possibly be detected by level gauges at a leak rate above 2 g/sec.

- 4588 EVALUATION OF THERMOCYCLIC STRENGTH OF STRUCTURAL MATERIALS AND STRUCTURAL ELEMENTS OF NUCLEAR POWER PLANTS WITH SODIUM HEAT CARRIER. Balandin, Yu.F.; Malygin, A.F.; Lisenkov, A.A.; Sennikov, E.V. Probl. Proch.; No. 3, 13-20 (Mar 1977). (In Russian).

A method of evaluation of the tensile properties is considered. The method is extended to parts of equipment made of structurally stable materials and subjected to the effects of thermal cycling and external loading. The theoretical evaluation of the strength of the structural elements in transient conditions is based on the utilization of experimentally determined limiting characteristics of resistance to thermal fatigue and long-time static failure. The proposed method is supported by the results of setup tests on half natural-size models in a sodium environment. 33 refs.

- 4589 MAIN TRENDS OF ACTIVITIES OF THE CMEA STANDING COMMISSION FOR PEACEFUL USES OF ATOMIC ENERGY. Neumann, J. (Ceskoslovenska Komise pro Atomovou Energii, Prague). Rev. Prum. Obchodu; 2: No. 3, 26-27 (Mar 1977). (In Czech).

The main spheres of activities are indicated of the Czechoslovak nuclear programme within COMECON, viz.: (1) The construction of more WWR-440 nuclear power plants - the V-2 power plant at Jaslovske Bohunice and the V-3 and V-4 nuclear power plants at Dukovany on the Jihlava river. (2) The Czechoslovak metallurgical and engineering industries start the production of reactor vessels, reactor internals, reactor control and protection systems, and the production of other components for the WWR-440 reactor primary circuits, i.e., steam generators, fittings, etc. (3) While continuing the production of WWR-440 reactor components the industries will gradually be introducing the production of WWR-1000 components. (4) The participation, with the other COMECON countries, in the research, development and manufacture of sodium cooled fast reactors. (5) The introduction of the production of nuclear power and heating plants.

- 4590 THERMOCYCLIC STRENGTH ESTIMATION FOR STRUCTURAL MATERIALS AND COMPONENTS OF NUCLEAR POWER PLANTS WITH SODIUM-COOLED REACTORS. Balandin, Yu.F.; Malygin, A.F.; Lisenkov, A.A.; Sennikov, E.V. Probl. Proch.; No. 3, 13-20 (Mar 1977). (In Russian).

A method for estimating the thermal-cycling resistance of elements of nuclear power stations with a sodium coolant was considered. The method extends to parts made of structure-stable materials and subjected to thermal

cycling and external loading. The calculated estimate of the resistance of the structural elements in transition regimes is based on the use of experimentally determined limiting characteristics of the resistance to thermal fatigue and to long-term static failure. The proposed method was proved by the results of stand testing of seminatural-size models in a sodium medium.

- 4591 SYSTEM FOR THE TRANSFER OF NUCLEAR FUELS OUTSIDE THE VESSEL. Wade, E.E. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). French Patent 2,343,314/A/. 23 Feb 1977. Priority date 23 Feb 1976, United States of America (USA), 19p. (In French).

The invention concerns a fuel transfer system for a LMFBF type reactor, for moving nuclear fuel assemblies between the reactor vessel and a storage container. The system has a fuel transfer chamber in communication with the reactor vessel and the storage container, and the storage container has an outlet orifice located outside the vessel, providing a storage area for the fuel assemblies. An out-of-vessel transfer system is located outside the reactor vessel and operates in conjunction with the storage container and the fuel transfer chamber, in order to transfer the fuel assemblies between the fuel transfer chamber and the storage container without placing the fuel assembly in a horizontal position. The transfer chamber and the storage container are generally placed below the level of the coolant fluid in the reactor vessel, so that the fuel assemblies stay completely immersed under a sheet of liquid coolant (sodium).

- 4592 FACILITIES OF FUEL TRANSFER FOR NUCLEAR REACTORS. Wade, E.E. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). French Patent 2,341,920/A/. 18 Feb 1977. Priority date 23 Feb 1976, United States of America (USA), 14p. (In French).

This invention relates to sodium cooled fast breeder reactors. It particularly concerns facilities for the transfer of fuel assemblies between the reactor core and a fuel transfer area. The installation is simple in construction and enables a relatively small vessel to be used. In greater detail, the invention includes a vessel with a head, fuel assemblies housed in this vessel, and an inlet and outlet for the coolant covering these fuel assemblies. The reactor has a fuel transfer area in communication with this vessel and gear inside the vessel for the transfer of these fuel assemblies. These facilities are borne by the vessel head and serve to transfer the fuel assemblies from the vessel to the transfer area; while leaving the fuel assemblies completely immersed in a continuous mass of coolant. A passageway is provided between the vessel and this transfer area for the fuel assemblies. Facilities are provided for closing off this passageway so that the inside of the reactor vessel may be isolated as desired from this fuel transfer area while the reactor is operating.

- 4593 DEVICE FOR MEASURING LIQUID SODIUM ABSOLUTE AND DIFFERENTIAL PRESSURES IN THE FAST NUCLEAR REACTOR COOLANT CIRCUIT. Bar, J.; Jung, J.; Pololanik, L. Czech Patent 167,817/B/. 15 Feb 1977. 3p. (In Czech).

A system with a vessel forming part of the circuit is proposed for measuring the absolute and the differential liquid sodium pressures in the fast reactor coolant circuit. The space above the sodium level in the vessel is filled with an inert argon atmosphere. Mounted onto

one vessel wall below the sodium level is a metal diaphragm closing one end of a tube filled with silicon oil. The other end of the tube is closed with a second metal diaphragm connected to the manometer pick-up.

- 4594 HEAT SHIELDS, FOR PUMP OR PIPE INSTALLATIONS. (to Neratoom N.V., The Hague (Netherlands)). Netherlands Patent 7,701,481/A/. 11 Feb 1977. 6p. (In Dutch).

1) The heat shield is characterised by the installation of at least two annular or plate-form components, so that during the operating process on optimal spacing is obtained between the shield and the wall surface. The heat shield is characterised by at least two layers of material with different coefficients of expansion. The heat shield has corrugations, two parts of sheet form wherein the mutual spacing is formed by means of a mechanical hinged link means, and a plate shaped part with an annular circumference so that on it the plate may be mounted and at least at one place may be sawcut and at least part of it may be of bimetallic form.

- 4595 PROCESS FOR DETECTING AND LOCATING FAILED FUEL ELEMENTS IN A NUCLEAR REACTOR. Wonn, J.W. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). French Patent 2,341,182 /A/. 9 Feb 1977. Priority date 11 Feb 1976, United States of America (USA), 17p. (In French).

The description is given of a process for detecting and locating a failed fuel element of a liquid cooled reactor. The fuel element includes fissile matter enclosed in a tubular metal cladding hermetically sealed both ends. The temperature of the fuel element is raised above the boiling point of the coolant so as to cause any liquid to boil that might enter the fuel element and the audio-frequency noise generated by the boiling liquid in the fuel element is monitored as the indication of a leak in the cladding.

- 4596 EXCHANGER PUMP UNIT FOR NUCLEAR REACTORS. Robin, M. (to CEA, 75 - Paris (France)). French Patent 2,379,881/A/. 4 Feb 1977. 36p. (In French).

This invention concerns an exchanger-pump unit comprising a double wall tube heat exchanger and a pump for circulating the primary exchange fluid integrated with this exchanger. This exchanger also enables any leaks through the walls of the exchange tubes to be monitored. The invention relates to a heat exchanger between two coolant fluids, primary and secondary respectively, which enables action to be taken on the exchange appliance, in the event of trouble, before the two fluids come into contact. This particularly concerns, although not exclusively the case, where the two fluids react chemically on each other and produce a destructive reaction. As will be imagined, special precautions have to be taken in such a case. Further, the apparatus described in the invention concerns the integration of the pump in the heat exchanger. This problem occurs particularly, although not exclusively, in the case of cooling systems of fast reactors cooled by liquid sodium or its alloys.

- 4597 METHOD FOR HEAT EXCHANGER LEAK DETECTION IN A LIQUID METAL COOLED NUCLEAR REACTOR. Sayre, R.K. (to Westinghouse Electric Corp.). US Patent T955,004. 1 Feb 1977. Filed date 5 Apr 1976. 2p.

An on-line method is described for early

detection of a leak through a tube of a liquid metal-to-water tube type heat exchanger in a liquid metal cooled nuclear reactor system. Introducing helium to the heat exchanger feedwater exposes the helium to any heat exchanger leak locations, and the helium will diffuse or flow through the leak location into the liquid metal. An increase in the helium content of the liquid metal evidences the existence of a leak. The content of helium in the liquid metal can be detected either by direct analysis of the reactor cover gas in a dual cycle reactor plant, or by removing a coolant sample from the reactor system downstream of the heat exchanger in a dual or triple cycle plant, separating the gaseous content of the sample, and analyzing the gaseous content for helium. The properties of helium are such that the invention method may result in identification of a tube leak before any water and liquid metal actually come in contact.

- 4598 STUDYING BEHAVIOUR OF HOT WASTES FROM REPROCESSING OF FAST REACTOR FUEL ELEMENTS BY THE GAS-FLUORIDE METHOD. Kirillovich, A.P.; Lavrinovich, Y.G.; Skiba, O.V.; Vorobei, M.P.; Starozhukov, D.I. At. Energ. (USSR); 42: No. 2, 94-98(Feb 1977). (In Russian).

The results of investigating certain physico-chemical and radiation properties of high level radioactive solid waste are presented. Reprocessing facility characteristics and investigation methods are described. Results of measuring heat generation from irradiated fuel and radioactive waste are reported. A fission product balance is presented. The radiation gas release rate is evaluated on the basis of observations over a period of one year. The amount of radionuclides released in the gas phase from solid waste is determined. The level of gas phase contamination during the waste storage is calculated. It is concluded that the radioactive waste should be placed in hermetically sealed containers for long-term storage under controlled conditions.

- 4599 STEAM GENERATORS OF SODIUM COOLED FAST REACTORS. Suransky, F. Jad. Energ.; 23: No. 2, 67-72(Feb 1977). (In Czech).

The basic classification of sodium-water steam generators by design is given, i.e., into body generators and modular (section) generators; the design of individual types is described and advantages and disadvantages listed. The individual systems differ also in operation, in the configuration of pipes and the design of the gas cushion. The basic demand put on all these steam generators is the protection of the system against the hazards of small or large sodium-water reactions. The structural criterion considered is the rupture of both walls of one pipe as the initial event; the growth of microleaks is also considered. The possibilities are given of leaks detection using chemical or physical processes arising during the sodium-water reaction. The criteria are listed for the selection of materials for steam generators, and the effects of corrosion related to the quality of feed water are considered.

- 4600 SAND FILTERS AS HIGH EFFICIENCY AEROSOL FILTERS IN NUCLEAR PLANTS. Boehm, L. (Rheinisch-Westfälischer Technischer Ueberwachungs-Verein e.V., Essen (Germany, F.R.)); Jordan, J.; Schikarski, W. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). pp 661-670 of Proceedings. Seminar on high efficiency aerosol filtration in the nuclear industry.

Luxembourg; Commission of the European Communities (Feb 1977). (In German)

From Seminar on high efficiency aerosol filtration in the nuclear industry; Aix-en-Provence, France (23 Nov 1976).

See CONF-761145--.

In Liquid Metal Fast Breeder Reactors sodium is used as the cooling medium. Accidents at these nuclear installations may occur as a result of pipework leakage or major spills of radioactive sodium. To minimize the release of radioactivity to the environment in an accident case, a special filtration system is necessary. To date, nuclear installations with a possibility of sodium fires have been equipped with conventional glass-fibre filters. These filters have high retention efficiencies for chemically inert aerosols but do not resist aggressive sodium oxide aerosols. Sand bed filters have therefore been developed and tested for use in the ventilation extract system of the SNR-300. Filtration of aerosols by sand filters is widely practised, although no previous attempts to optimize them as regards efficiency and loading capacity have been reported. Experiments have been carried out with high aerosol concentrations (up to 40 g/m³) and flow velocities of 0-5 cm/sec in the Karlsruhe Nuclear Research Centre.

- 4601 HIGH EFFICIENCY FILTRATION OF LIQUID-METAL-GENERATED AEROSOLS. First, M.W. (Harvard School of Public Health, Boston, Mass. (USA)). pp 519-531 of Proceedings. Seminar on high efficiency aerosol filtration in the nuclear industry. Luxembourg; Commission of the European Communities (Feb 1977).

From Seminar on high efficiency aerosol filtration in the nuclear industry; Aix-en-Provence, France (23 Nov 1976).

See CONF-761145--.

This paper will present data on filter performance for the range of sodium-containing aerosols that can result from large and small releases of hot metallic sodium into confined spaces containing air or special atmospheres and will compare the particle collection effectiveness, space and power requirements, service life, and disposal of the collected materials for the several options. Prompt reduction of in-vessel aerosols with methods that induce rapid coagulation and sedimentation by the application of violent turbulence, sonic energy, or electrostatic attraction; by aerosol scavenging with massive inert dust additions; and by a number of other innovative methods is of special interest because of their potential ability to bring down the aerosol cloud very rapidly and thereby to reduce vessel out-leakage drastically; as well as to relieve the particle load on filters. These methods will be examined as supplements to filtration methods for control of sodium-containing aerosols.

- 4602 AVAILABILITY ANALYSIS OF COMMERCIAL FAST BREEDER REACTOR'S STEAM GENERATOR SYSTEM. Maekawa, I.; Shindo, Y.; Sakano, K. (Kawasaki Heavy Industries Ltd., Kobe (Japan)). FAPIG (Tokyo); No. 84, 8-12(Feb 1977). (In Japanese).

Optimization study for the availability and load factor of the sodium heated steam generator for a commercial fast breeder reactor was conducted as one of the economical feasibility study. Three cases were chosen for this availability analysis. The numbers of evaporator, super-heater and reheater are one each in case A, 4, 2 and 2 respectively in case B, and 5, 4 and 4 respectively in case C, and this plant has 1500 MWe capacity and four loops. The analysis was conducted by fixing the range of analysis, and then the factors of

plant outage especially for the probability of faults and maintenance time, and each fault mode was represented by the fault tree. The unavailability per each loop, the system unavailability and the load factor in cases A, B and C are shown, and the analytical results show that the load factor is the best in case B in this analysis. It is necessary to research the probability of faults in steam generator to accumulate the field data, and to investigate the system and the operational method to lessen low availability machinery to increase their availability.

- 4603 FILTRATION OF AEROSOLS PRODUCED BY A SODIUM FIRE. Duverger de Cuy, G.; Colome, J. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. de Surete Nucleaire). pp 533-544 of Seminar on high efficiency aerosol filtration in the nuclear industry. Luxembourg; Commission of the European Communities (Feb 1977). (In French)

From Seminar on high efficiency aerosol filtration in the nuclear industry; Aix-en-Provence, France (23 Nov 1976).

The containment system of the Super Phenix reactor takes account of the possibility of contaminated sodium fires, particularly in the vicinity of the fuel storage drum. It is thus necessary to contain the emitted radioactivity associated with a quantity of sodium aerosols of the order of some 10 g/m³. Investigations previously carried out had shown that the retention capacity of high-efficiency asbestos filters was clearly insufficient. A new research programme on the filtration of sodium aerosols has thus been worked out with the aim of: gaining a better understanding of the granulometry of the aerosols produced by the fire; checking the efficiency of the new glass-fibre media of which high-efficiency filters are composed; selecting a prefiltering system which, in conjunction with the high-efficiency filter, would ensure a suitable retention capacity for the whole unit. The following prefilterers are under investigation: agglomerating cyclones; fabric prefilterers (dense filter media - variable density filters - bag filters); water prefilterers; sand prefilterers. The experimental equipment on which this programme has been based is presented. Results have so far been obtained for agglomerating cyclones (recirculating loop with a cyclone of 95% efficiency), for a number of textile prefilterers and for the retention capacity of high-efficiency filters.

- 4604 (DOE-tr--179) TUFCEI, AN EXPERIMENTAL FACILITY TO STUDY THE MOLTEN-FUEL-COOLANT INTERACTION IN A FULL-SCALE FUEL ELEMENT. Beutel, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). Feb 1977. Translation of KFK--2353 65p. Dep. NTIS, PC A04/MF A01.

The safety investigations on the Na-cooled fast breeder reactors also include experiments on the fuel-sodium reaction, in which especially the conversion of thermal energy into mechanical energy is to be determined in a simulated fuel element. The development of a test facility is described in which fuel-sodium reactions will be carried out in geometries which are typical for fuel elements. In contradistinction to the actual fuel elements in a core reactor, the walls in the reaction room are almost rigid for the expected pressure transients. The sodium- UO_2 contact is then produced both by the sodium striking the molten UO_2 and by injection of small amounts of sodium into the UO_2 (initiation reaction).

- 4605 SCRAM ROD SYSTEM FOR NUCLEAR REACTORS. Golden, M.P.; Govi, A.R. (to Westinghouse Electric Corp., Pittsburgh, Pa. (USA)). French Patent 2,339,937/A/. 21 Jan 1977. Priority date 28 Jan 1976, United States of America (USA), 34p. (In French).

A system of safety rods developed for sodium cooled reactors is described. Each rod is connected to a drive rod encircled by concentric bellows tightened to a magnetic plate at their upper end. These bellows constitutive the driving means and the annular space between them is in relation with pressure gas inlet and outlet means. An electromagnetic device is fitted up at the upper end of the drive rod stroke. The drive rod is maintained out of the nuclear reactor by the combined action of the electromagnetic device and the reduced gas pressure between the bellows.

- 4606 FUEL ASSEMBLY. Nishi, H.; Yoshida, H. (to Toshiba Corp., Kawasaki, Kanagawa (Japan)). Japanese Patent 1953-90,587/A/. 18 Jan 1977. 3p. (In Japanese).

A method to minimize the changes in the temperature distribution due to deviation of fuel pins in fuel assemblies of an lmfbr type reactor is described. A fuel assembly is supported on a plurality of support plates provided at the lower inside of a wrapper tube. Fasteners at the lower end plug of cladding tubes for closing the lower end thereof are assembled in such way that the directions of the fasteners are opposed to each other between each of adjacent fuel pins, in particular, those fuel pins supported on different support plates. With the above constitution, if the fuel pins tend to move laterally in the same direction acted by fluid pressure due to coolant flow, deviation of the fuel pins is not resulted to all of the pins since at least one of the adjacent fuel pins is fixed to the support plate by the fastener to be hindered from displacement.

- 4607 SUPPORTS FOR PIPES. Pollono, L.P. (to Westinghouse Electric Corp., Pittsburgh, Pa. (USA)). French Patent 2,339,800/A/. 17 Jan 1977. Priority date 28 Jan 1976, United States of America (USA), 9p. (In French). Support means for hot thin-walled pipes used in cooling circuits of liquid metal cooled nuclear reactors are described. They consist of tightening means put around the pipe, and a supporting structure. An insulating annular means is set up between the pipe and the tightening means: it is hollow and contains thermal insulating materials.

- 4608 NEUTRON TRANSPORT STUDIES IN FAST REACTOR SHIELDS. Vaidyanathan, R.; Murty, K.P.N.; Singh, R.S. (Reactor Research Centre, Kalpakkam, India). Natl. Bur. Stand. (U.S.), Spec. Publ.; No. 461, 163-165 (Jan 1977).

From International symposium on radiation physics; Calcutta, India (30 Nov 1974). See CONF-741109--.

The present study aims at evaluating the relative accuracies with which the different approximations to the transport theory represent the spatial and spectral distribution of neutrons in a typical fast reactor shield. Diffusion, removal diffusion and discrete ordinate transport calculations, with and without the inclusion of anisotropic scattering effects, have been performed for the lateral and axial shield configurations of the French fast reactor Rapsodie. The results have been compared, with the measurements made in Rapsodie and the NIOBE calculations made

earlier.

- 4609 (EURFNR--1447) HEATING RODS FOR SIMULATION OF FUEL PINS IN LIQUID METAL BOILING EXPERIMENTS. PART I. Casal, V.; Demski, A.; Graf, E.; Hartmann, W. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorbauelemente). Jan 1977. Translation of KFK--2368 48p. Dep. NTIS, PC A03/MF A01.

Work performed under United States--Euratom Fast Reactor Exchange Program.

This report describes the design, development and testing of electrical heater rods for boiling tests in LMFBR fuel element simulators. The tests are required as a proof of the capability to control local and gross boiling in the core of SNR-300. The setup and performance of the tests call for heater rods simulating with a very high operating reliability the assumed incidents of SNR-300, even under transient conditions. Loads of the heating surface of up to 170 W/cm² must be attained at coolant temperatures of 950°C. The concept selected for the heater rod is an improvement of heater rods used in single-phase heat transfer tests. The heat is generated in the heater rod by ohmic heating of a tape helix. The heat passes the insulator, a layer made of compacted boron nitride powder, and reaches the cladding of the heater rod. The current conductor is made of a nickel-chromium alloy while the cladding material is stainless steel. The concept of the heater rod and the fabrication method have proved to be good. For instance, rejections in the fabrication process, including the instrumentation, are less than 1 percent. In the boiling experiments performed on two 7-rod bundles and one 169-rod bundle containing 88 heated rods no heater rod has failed so far.

- 4610 EVALUATION OF DESIRABLE ACCURACY FOR NUCLEAR DATA IN FAST REACTORS BY OPTIMIZATION METHOD. Miki, K.; Inoue, K. (Hitachi Ltd., Kawasaki, Kanagawa (Japan). Atomic Energy Research Lab.). J. Nucl. Sci. Technol. (Tokyo); 14: No. 1, 58-64 (Jan 1977).

The desirable accuracy of nuclear data for minimizing the design margin in fast reactor nuclear designs is evaluated through an optimization method, in which a parameter termed the "degree of difficulty" has been introduced in order to represent the relative degree of difficulty expected to be encountered in improving the accuracy of the data. This parameter serves to determine the desirable accuracy, by which the design margin could be reduced the least onerously, to obviate uselessly severe requirements being imposed on the accuracy demanded of data with high sensitivity coefficients. Application of this method to the equilibrium core of a prototype fast reactor leads to the conclusion that nuclear data that should to best advantage be improved in accuracy are: $\sigma_{sub(f)sup(41)}(400 \text{ -- } 0.5\text{keV})$, $\sigma_{sub(f)sup(49)}(1.4\text{MeV -- } 10\text{keV})$ and $\sigma_{sub(c)sup(28)}(100 \text{ -- } 0.5\text{keV})$. Furthermore, it is indicated that the ratio between the desirable accuracy and the present uncertainty level nuclear data is roughly inversely proportional to the square root of the product of the sensitivity coefficients and the present uncertainty level of nuclear data.

- 4611 PHENIX INTERMEDIARY HEAT EXCHANGER INTERVENTION. Conte, F. (CEA Centre de Marcoule, 30 - Bagnols-sur-Cèze (France). Service de la Centrale Phenix). Rev. Gen. Nucl.; No. 1, 57-59 (Jan-Feb 1977). (In French).

Operation on Phenix intermediary heat exchangers after the July and October 1976 incidents is analyzed. Included are remarks on conditions of repair work.

- 4612 HEAT TRANSFER CHARACTERISTICS OF STEAM GENERATORS FOR FBR. Takahashi, T. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center); Tashino, M.; Sano, T.; Munezane, S.; Kiyokawa, T. Mitsubishi Juko Giho; 14: No. 1, 39-52 (Jan 1977). (In Japanese).

The development of the Fast Breeder Reactor (FBR) is urged for the effective use of the uranium resources. The Japanese Government has launched a national program plan to develop the demonstration FBR plant 'MONJU'. The Power Reactors and Nuclear Fuel Development Corporation (PNC) is charged with the responsibility for this program and also has been supervising it. The Steam Generator (SG) is one of the most important component in FBR plant. The high reliability is required in design, manufacture and installation of the SG because it exchanges the heat from sodium to water/steam. For these reason, the development work of the SG has been one of the critical area in the national program plan and consisted of the following three phases. The first one was for small scale 1 MW SGs and was already over. The second one was for medium scale 50 MW SGs and the last one is for 'MONJU' 238 MW SGs. All these SGs are the so-called helically coiled tube type units. Mitsubishi Heavy Industries received an order of the 50 MW SG No. 2 from PNC. Through this 50 MW SG No. 2 work, the various studies and experiments were done to design and manufacture the unit. The unit was completed and installed in Oct., 1975. Since then, PNC has been operating and testing it successfully. It is very important to examine the operational and test data from various technical aspects in every phase in order to confirm the design. This paper presents the thermal performance of 50 MW SGs comparing with its test data.

- 4613 OUTLINE OF 'MONJU' COORDINATED DESIGN, (3). Doryokuro Giho; No. 21, 53-78 (Jan 1977). (In Japanese).

Design of 'Monju', which is planned as a power-generating plant using the fast breeder reactor prototype, progressed in order. The coordinated design (3) was performed from November '75 to June '76. Based on the coordinated design (2), more minute machine design, unification of the specifications, the safety analysis, the aseismatic design, and the preparation of a variety of design criteria were achieved. The reactor auxiliary building was made a little smaller than in the coordinated design (2) by coordinating the arrangement of the equipments. Installation schedule and procedures were discussed. The reactor core has nearly cylindrical shape, and is composed of 198 core fuel assemblies and 172 blanket fuel assemblies. Hereinafter the following items are described: reactor core, reactor structure, fuel-handling system, heat-transferring system, steam-generating system, turbo-generator system, plant operation mode, electrical, instrumentation and control facilities, radioactive waste disposal system, and finally the reactor container and building, and miscellaneous facilities. The miscellaneous facilities include sodium supply, argon gas supply, nitrogen gas supply, sampling and analyzing, compressed air supply, auxiliary machine cooling, fresh water supply, fire extinguishing, maintenance and air conditioning systems.

- 4614 PRE-HEATER FOR FAST BREEDER REACTOR AND ITS LIFE. Takahashi, Y. (Sukegawa Electric Co. Ltd., Tokyo (Japan)). Haikan Gijutsu; 19: No. 1, 96-108 (Jan 1977). (In Japanese).

Pre-heating must be performed at the beginning of operation of the nuclear equipments using liquid metal as a heating medium, such as fast breeder reactors. The simplest means is to employ electric heaters. The life term of heaters has been tested because the life term same as that of reactors themselves has been required for the heaters. The heaters for pre-heating sodium facilities include sheathed heaters and micro-heaters, both of which have the construction that the heating wires are inserted in metal tubes filled with magnesia. After theoretically discussing the temperature difference and thermal stress of sheaths and heating wires, temperature cycle and stress cycle, low cycle thermal fatigue at high temperature and the theory of L.F. Coffin are explained, and the experiments have been performed based on the Coffin's theory. There were suspicions that inorganic heat insulators further fire as the water supplier to sodium and corrode stainless steel because of containing chlorides. Though it has been recently considered better to use metal insulation (utilizing the reflective index of polished aluminum surface), here the heat stored in inorganic heat insulators is calculated, and the conditions of heat conduction and heater restraint are experimented. In addition, the insulation resistance of heaters, the material, pitch non-uniformity, and eccentricity of heating wires are also discussed. At present, conclusion can not be obtained simply.

- 4615 (EURFNR--1425) EXPERIMENTAL INVESTIGATIONS INTO COOLANT CROSS MIXING IN MULTI-ROD CLUSTERS WITH GRID-TYPE SPACERS IN TURBULENT SODIUM FLOW. Hoffman, H.; Baumgartner, E.; Frey, H.H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorbauelemente). Jan 1977. Translation of KFK--2360 36p. Dep. NTIS, PC A03/MF A01.

Work performed under United States--Euratom Fast Reactor Exchange Program.

The thermo- and fluid-dynamic design of fuel elements of Liquid Metal-Cooled Fast Breeder Reactors needs characteristic values for the magnitudes of the coolant cross mixing. These were determined experimentally in sodium flow. 61-rod clusters were used with the specific dimensions of the SNR-300 fuel element (rod pitch to diameter ratio $P/D = 1.32$) equipped with grip-type spacers. The investigations were performed using the "heating method." Parameters are: the geometry of spacers (honeycomb-, ferrule-type); the Reynolds-numbers (8×10^3 less than or equal to Re less than or equal to 8×10^4); the support length of the rods within the cluster (150 less than or equal to H /mm/ less than or equal to 450); and the distance between grids and measuring plane (50 less than or equal to Δh /mm/ less than or equal to 160). In addition experiments were carried out using a 61-rod-cluster, $P/D = 1.2$, equipped with honeycomb-grids. Measured temperature profiles within the coolant outlet cross-section are reported as well as the mixing rates evaluated with respect to its definition in the MISTRAL-II-Code. The results show that the mixing rates are mainly influenced by the spacer type and the P/D -ratio.

- 4616 DEVELOPMENT OF STEEL TUBES FOR STEAM GENERATOR OF FAST BREEDER REACTOR. Yukitoshi,

T.; Moroishi, T.; Koizumi, I.; Yoshikawa, K.; Shida, Y. (Sumitomo Metal Industries Ltd., Amagasaki, Hyogo (Japan). Central Research Labs.). Sumitomo Kinzoku; 29: No. 1, 32-42 (Jan 1977). (In Japanese).

When austenitic steel and ferrite steel coexist in liquid sodium, the elements in the steels leach out and move through the liquid sodium. Especially when the decarburization occurs in ferritic steel, the lowering of high temperature strength becomes a design problem. The high-temperature strength, weldability, and decarburization behavior of 2 1/4 Cr-1 Mo steel with a wide variety of heat treatments and some carbon-stabilized 2 1/4 Mo steels in liquid sodium were examined for the purpose of material selection for the steam generator tubes of liquid-metal fast breeder reactors (LMFBR). The decarburization of 2 1/4 Cr-1 Mo steel in sodium depends on the matrix and the precipitates, and can be suppressed when the steel is appropriately normalized and tempered. The 2 1/4 Cr-1 Mo-Nb-Ni steel is superior to other stabilized steels, since the small addition of nickel assures sufficient strength, even if the normalizing temperature is as low as 950 degree C. Stabilized steels should be welded carefully to obtain fully penetrating beads. Preheating above 200 degree C and postheating above 720 degree C are required for the stabilization. As comparing all the investigated properties, the materials recommended for LMFBR steam generator tubes are normalized and tempered 2 1/4 Cr-1 Mo steel and, when the decarburization is not allowable, 2 1/4 Cr-1 Mo-Nb-Ni steel, with the condition that sound-welded joints can be available.

- 4617 REACTIVITY EFFECTS IN THE BN-350 REACTOR. Drov, V.V.; Troyanov, M.F.; Matveev, V.I. (and others). At. Energ. (USSR); 42: No. 1, 3-8 (Jan 1977). (In Russian).

The results of theoretical and experimental investigations of reactivity coefficients of the LMFBR type BN-350 reactor are presented. The isothermal temperature reactivity coefficients have been measured at temperatures ranging from 180 to 280 deg C. The corresponding calculated values take into account temperature induced changes in the geometry of the core, resonance neutron cross sections (the Doppler effect) and the coolant density (sodium). The power reactivity coefficients have been measured during step-by-step increase of the core's rating. The Doppler effect is noted to mostly contribute to the changes in the power reactivity coefficients (approximately 70%). The reactivity effects connected with variations in both the breeding ratio and the isotopic composition of the fuel have been also measured. It is pointed out that the experiments performed are the first-of-a-kind verification of the effects of temperature, power and fuel burn up on reactivity of a power fast breeder reactor. The models and methods used to calculate and design power fast breeder reactors are stressed to have proved valid and quite satisfactory.

- 4618 (BNWL-tr--340) RADIOACTIVITY OF LONG-LIVED NUCLIDES IN THE PRIMARY LOOP OF THE BOR-60 REACTOR DURING OPERATION WITH DEFECTIVE FUEL ELEMENTS. Gryazev, V.M.; Kizin, V.D.; Lisitsyn, E.S.; Polyakov, V.I.; Chechetkin, Yu.V. Jan 1977. Translation of NIIAR-P-5, No. 299, January 1977 23p. Dep. NTIS, PC A02/MF A01.

Generalized measured results are presented on the accumulation and distribution of radioactive nuclides from corrosion and fission products during the four years of operation of the BOR-60, including fairly long operation

with defective fuel elements in the core.

- 4619 ACOUSTICALLY INDUCED VIBRATION OF CIRCULAR CYLINDRICAL RODS. Lin, H.C.; Chen, S.S. (Argonne National Lab., IL). J. Sound Vib.; 51: No. 1, 89-96(1977).

An analysis is presented for the steady-state response of a circular cylindrical rod subjected to a plane wave; expressions for the rod displacement, the natural frequency (including the fluid inertia effect) and the radiation damping are obtained in closed form. It is shown that the added mass coefficient depends only on the non-dimensional wave number α and the radiation damping depends on α and the mass ratio β . The rod displacement is proportional to wave amplitude and rod radius, and depends on α and β .

- 4620 HOLLOW SPHERE FORMATION IN A GASSY TWO-LIQUID POOL. Bankoff, S.G.; Horin, W.A. (Northwestern Univ., Evanston, IL). Lett. Heat Mass Transfer; 4: No. 3, 225-229(1977).

Molten fuel pools with an overlying layer of steel can be expected to form in various stages of the LMFBR post-accident heat removal sequence, either on in-vessel or ex-vessel core catchers, or on a concrete or refractory floor beneath the reactor vessel. Since the boiling point of steel is approximately the same as the freezing point of UO_2 , one can expect that a crust of frozen UO_2 may form at the interface. However, there may be significant bubbling through the interface, either by boiling within the fuel pool or by gas released from the concrete or sacrificial material. The original objective of this work was therefore to study the conditions for crust formation in a simulant (hexane and water) pool with gas evolution. A unique physical phenomenon was observed, however, consisting of the formation of "ping-pong" balls of ice formed by water transported by the bubbles entering the cold overlying hexane. These formed a raft of hollow spheres at the top of the hexane layer. By analogy, one might expect considerable mixing of UO_2 in the form of hollow spheres into the molten steel, with important consequences for the pool geometry and heat transfer.

- 4621 ROLE OF NUCLEAR POWER IN THE NEXT FEW DECADES. Esselman, W.H. (Electric Power Research Inst., Palo Alto, CA). pp 16p, Paper 7 of Executive conference on uranium fuel supply. Hinsdale, IL; American Nuclear Society (1977).

From Conference of the American Nuclear Society; Monterey, CA, USA (24 Jan 1977). See CONF-770109--.

This paper presents a perspective on the potential of nuclear power over the next 25 years, based on EPRI studies. Factors affecting the electricity requirement are discussed; for the year 2000, it is projected to be 2.5 to 5 times the 1976 electricity usage. This increase will be met by coal and nuclear. It is doubtful that a large number of LMFBRs will be operating by 2000, and the potential energy resource in the spent LWR fuel elements is pointed out. (DLC)

- 4622 PREDICTION OF THE THERMAL CONDUCTIVITY OF UO_2 VAPOR. Tills, J.L.; Cronenberg, A.W. (New Mexico Univ., Albuquerque (USA)). Dept. of Nuclear Engineering; Schmidt, T.R. (Sandia Labs., Albuquerque, N.Mex. (USA)). J. Nucl. Mater.; 67: No. 1, 67-76(1977).

An important aspect of Liquid-Metal Fast Breeder Reactor (LMFBR) safety analysis is an accurate description of the transport phenomena

of the various reactor materials during a hypothetical core-disruptive accident. For the case of an accident sequence leading to fuel vaporization, the problem of energy losses by conduction, radiation, and latent heat and mass losses by condensation of the fuel vapor in the presence of cold structure and/or coolant is of particular interest since such losses will determine, to a large extent, the energetics of the fuel vapor expansion process following pin failure. In this paper, a theoretical estimate of the thermal conductivity of UO_2 vapor is determined, based upon kinetic theory for a multicomponent gas. Results indicate that at elevated temperatures, the conductivity is a strong function of temperature due to thermal dissociation of the UO_2 molecule, where the species concentrations are taken from calculations presented elsewhere. At 5000 K the thermal conductivity is 2.2×10^{-3} cal/cm-sec K, which is relatively high; thus, it should be considered in analyses of fuel vapor phenomenon. The functional relationship between conductivity and temperature can be approximated as: $\lambda_{\text{sub}}(T) = \exp[-39.66 + 1.891(-2)T - 3.561(-6)T^2 + 2.242(-10)T^3]$, for $3000 \text{ K} \leq T \leq 5500 \text{ K}$.

- 4623 ENERGY POLICY AND THE ROLE OF NUCLEAR POWER. Brookes, L.G. (UKAEA). pp 2.1-2.11 of Proceedings of symposium on energy in the 80s. London; Institution of Chemical Engineers (1977).

From Symposium on energy in the 80s; Middlesborough, UK (5 - 7 Apr 1977).

It is shown that sharply increased oil prices have harmed GNP performance and severely reduced the rate of substitution of nuclear for fossil energy. Reduction or stabilization of energy prices should be the primary objective of energy policy. It is suggested that the UK could help to maintain economic momentum by making the maximum use of gas while exporting oil. The taxes and royalties from the oil might best be fed back into the economy by reducing other forms of taxation. The complex interaction of the costs and prices of different fuels is discussed. The role of nuclear energy is considered. It is concluded that it is the cheapest, safest and cleanest way of producing electricity, and it is suggested that it should be bound up with the exploitation of the fast breeder reactor. The role of coal and of the Middle East oil reserves themselves is also considered.

- 4624 WAKE AND DISPERSION MODELS FOR THE EBR-II BUILDING COMPLEX. Halitsky, J. Atmos. Environ.; 11: No. 7, 577-596(1977).

Wind properties measured downwind of the reactor containment structure at the EBR-II complex were found to be in agreement with predictions made with a mathematical wake model that was derived from measurements in the wakes of suspended flat plates in a wind tunnel and modified for the presence of the ground boundary. Simultaneous concentration measurements of a tracer released in the lee of the reactor containment structure were found to be in agreement with the predictions of a mathematical dispersion model that incorporated the wake model in conjunction with atmospheric turbulence properties. The analysis provides insight into the nature and properties of building wakes in the atmosphere, and their influence on dispersion of released material.

- 4625 FLOW MODEL TEST DEVELOPMENT OF CRBRP INTERMEDIATE HEAT EXCHANGER. AbuRomia, M.M.; Vosch, R.J. Jr.; Cho, S.M.; Dietz, D.; Jaisingh, G. (Foster Wheeler Dev Corp,

- Livingston, NJ). Am. Soc. Mech. Eng., [Pap.]; No. JPGC-NE-10, 1-11(1977).
- The details of these tests, flow data achieved, finalized design configurations of the flow distribution devices, and final interpretation of the results are given in this paper. The tests consist of the: Primary Inlet Plenum Flow Model Test which has the objective of finalizing the design of the primary flow inlet plenum configuration that results in a circumferentially uniform flow entering the tube bundle; Tube Bundle Flow Model Test which has the objective of achieving optimum design configurations for the flow baffles in the bundle, which is based on the baffles capability to create the necessary forced fluid mixing of the shell side flow within the pressure drop limitation imposed on the unit's design; and Intermediate Flow Model Test with its objective of confirming the design of the intermediate flow inlet plenum, which has a hemispherical shape with a ring baffle attached, in distributing the flow evenly among the tubes. The paper includes details of the flow fields within various test models in a wide range of operating flow rates. 2 refs.
- 4626 STUDY OF THE DENSITY INTERFACE IN A STRATIFIED LMFBR OUTLET PLENUM. Lorenz, J.J.; Howard, P.A. (Argonne Natl Lab, Ill). Am. Soc. Mech. Eng., [Pap.]; No. 77-HT-23, 1-8(1977).
- Tests were performed with small scale model plenums using water as the test fluid. The movement of the interface was detected by vertical strings of closely spaced thermocouples. Considerable wave motion and unsteadiness were observed at the density interface. As the interface passed a given thermocouple location, the temperature was observed to drop abruptly while oscillating in the process. Using the theory of free submerged jets, a correlation was developed for the rise rate of the interface in terms of the Froude number based on local jet impingement parameters. This correlation can readily be incorporated into a computer code for outlet plenum analysis. 12 refs.
- 4627 CRBRP OUTLET PLENUM BEHAVIOR DURING STEADY STATE AND TRANSIENT CONDITIONS. Howard, P.A.; Lorenz, J.J.; Howarth, W.L.; Novodstern, E.H. (Argonne Natl Lab, Ill). Am. Soc. Mech. Eng., [Pap.]; No. 77-HT-32, 1-8(1977).
- Of major concern in this study were buoyancy-induced flow stratification and thermal oscillations ("thermal striping"). Tests were performed with a 1/10 scale model CRBRP outlet plenum complete with the Upper Internals Structure (UIS) using water as the test fluid. The components in the upper region of the outlet plenum experienced the characteristic temperature drop and recovery during the early development of flow stratification; temperature oscillations were observed at the stratification interface and in the neighborhood of the fuel assembly/blanket assembly boundary. Temperature measurements obtained provided the thermal boundary conditions required for calculating thermal stresses. 7 refs.
- 4628 VALIDATION OF TURBULENCE MODELS FOR LMFBR OUTLET PLENUM FLOWS. Chen, Y.B.; Golay, M.W. (MIT, Cambridge, Mass). Am. Soc. Mech. Eng., [Pap.]; No. 77-HT-34, 1-8(1977).
- Small scale experiments involving water flows are used to provide mean flow and turbulence field data for LMFBR outlet plenum flows. Measurements are performed at Reynolds number (Re) values of 33000 and 70000 in a 1/15 - scale FFTF geometry and at Re = 35000 in a 3/80-scale CRBR geometry. The experimental behavior is predicted using two different two-equation turbulence model computer programs, TEACH-T and VARR-II. It is found that the qualitative nature of the flow field within the plenum depends strongly upon the distribution of the mean inlet flow field, importantly also upon the degree of inlet turbulence, and also upon the turbulent momentum exchange model used in the calculations. In the FFTF geometry, the TEACH-T predictions agree well with the experiments. 7 refs.
- 4629 COMBINED FORCED AND FREE LAMINAR CONVECTION IN VERTICAL ROD BUNDLES WITH LONGITUDINAL FLOW. Ramm, H.; Johannsen, K. (Inst fuer Kerntechnik, Berlin, Ger). Am. Soc. Mech. Eng., [Pap.]; No. 77-HT-44, 1-12(1977).
- The heat-transfer problem of combined forced and free convection by fully developed laminar flow in a vertically mounted reactor fuel rod assembly has been solved analytically for both upflow and downflow. Results reveal the strong augmentation of buoyancy effects by radial gradients in the power generation across the assembly. 12 refs.
- 4630 ECONOMETRIC ANALYSIS OF ENERGY OVER THE NEXT 75 YEARS. Schmitt, R.W.; BenDaniel, D.J.; Stewart, P.J.; Manne, A.S. (GE Co, Schenectady, NY). IEEE Trans. Power Appar. Syst.; PAS-96: No. 4, 1353-1361(1977).
- An econometric model with a 75-year time horizon to determine base-load power generation mix is described. A number of conventional, evolutionary, and speculative generation systems are considered, together with constraints imposed by fuel availability and technical development times. Preliminary results are presented. 6 refs.
- 4631 ELECTROMAGNETIC FLOWMETERS FOR FAST REACTORS. Baker, R.C. (Imperial Coll. of Science and Technology, London (UK). Dept. of Mechanical Engineering). Prog. Sci.; 1: No. 1, 41-61(1977).
- This paper is in four parts. Section 1 explains the theory of the induced-voltage electromagnetic flowmeter and then considers various types which have been used. For the primary circuit of fast reactors both flow-through type and probe type have been proposed, although obtaining magnets which operate satisfactorily at high temperatures has been a problem. In the secondary circuit the high magnetic Reynolds numbers cause the field to be swept out of the magnet gap and this has led to the use of the long saddle-coil flowmeter. In Section 2 flux-distortion flowmeters are described. These have been proposed mainly for monitoring the primary circuit flow and again both flow-through and probe types have been tested. Sections 3 and 4 continue the discussion of the flux-distortion flowmeter by introducing two methods of analysing its performance. The first is a finite difference method which solves the non-linear problem by using a time marching method. It is shown that a linear approximation is adequate for the likely levels of flow encountered in the fast reactor and consequently two linearised solutions are used. The first method is a finite difference one and allows the instantaneous response of a step change in velocity to be observed as well as the effect of bubbles. In Part 4 the second linearized method uses current rings to divide up the conducting material. By considering the

interaction of all the rings, it is possible to obtain the current distribution and hence the magnetic field. In conclusion it is suggested that further development would be useful of the devices which are most suited to the liquid metal fast breeder reactor.

- 4632 COPROCESSING TO AMELIORATE THE PROLIFERATION AND WASTE MANAGEMENT PROBLEMS. Pobereskin, M.; Kok, K.; Madia, W.J. (Battelle Columbus Labs., OH). Trans. Am. Nucl. Soc.; 27: 428-429(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

SPENT FUELS;PROLIFERATION;RADIOACTIVE WASTE MANAGEMENT;REPROCESSING;LMFBR TYPE REACTORS;PWR TYPE REACTORS;BWR TYPE REACTORS;ENVIRONMENTAL EFFECTS

- 4633 DESIGN AND LICENSING OF A FLEXIBLE DUAL-CONTAINMENT APPROACH. Kent, G.F.; Willie, D.J. (Stone and Webster Engineering Corp., Boston). Trans. Am. Nucl. Soc.; 27: 645-646(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

PWR, BWR, LMFBR. PWR TYPE REACTORS;BWR TYPE REACTORS;LMFBR TYPE REACTORS;CONTAINMENT BUILDINGS;SPECIFICATIONS

- 4634 VAPOR FILM COLLAPSE DURING SUBCOOLED FILM BOILING AND ITS APPLICATION TO MOLTEN METAL COOLANT INTERACTIONS. Purohit, G.P. (Jet Propulsion Lab., Pasadena, CA); Dhir, V.K. Trans. Am. Nucl. Soc.; 27: 665-666(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

BWR, PWR, LMFBR. PWR TYPE REACTORS;BWR TYPE REACTORS;LMFBR TYPE REACTORS;MELTDOWN;FUEL-COOLANT INTERACTIONS;BOILING;HEAT TRANSFER; TEMPERATURE GRADIENTS;MOLTEN METAL-WATER REACTIONS

- 4635 BARYCENTRIC COORDINATES TECHNIQUE FOR RESOLUTION OF MULTIPLE FUEL FAILURES WITH GAS TAGGING. Gross, K.C. (Argonne National Lab., IL); Passerello, C.; Shapiro, A. Trans. Am. Nucl. Soc.; 27: 685-686(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

PWR, BWR, LMFBR. PWR TYPE REACTORS;BWR TYPE REACTORS;LMFBR TYPE REACTORS;FAILED ELEMENT MONITORS;FAILED ELEMENT DETECTION;FUEL ELEMENT FAILURE;THREE-DIMENSIONAL CALCULATIONS;FINITE ELEMENT METHOD

- 4636 PROBABILISTIC IDENTIFICATION OF FAILED CORE ASSEMBLIES FROM GAS TAG RELEASES. DeClue, J.F. (Univ. of Washington, Seattle); McCormick, N.J.; Schmittroth, F. Trans. Am. Nucl. Soc.; 27: 686-687(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

PWR, BWR, LMFBR. PWR TYPE REACTORS;BWR TYPE REACTORS;LMFBR TYPE REACTORS;FAILED ELEMENT MONITORS;FAILED ELEMENT DETECTION;FUEL ELEMENT FAILURE;PROBABILITY

- 4637 IMPLICIT EULERIAN HYDRODYNAMICS TECHNIQUE FOR MULTICOMPONENT SYSTEMS. Weber, D.P. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 383-384(1977).

From ANS winter meeting; San Francisco, CA,

USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMFBR TYPE REACTORS;LOSS OF FLOW; HYDRAULICS;HYDRODYNAMICS;MATHEMATICAL MODELS

- 4638 COMPARISON OF THE PERFORMANCE OF SEVERAL NEUTRON DIFFUSION COMPUTER CODES FOR A FAST BREEDER REACTOR. Kuefner, K.; Buckel, G.; Stehle, B. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). Atomkernenergie; 30: No. 2, 77-81(1977).

From Technical meeting on coarse-mesh methods: progress in the methods and their application to reactor problems; Erlangen, Germany, F.R. (25 - 26 Jan 1977).

Calculations have been performed with four computer codes solving the static multigroup neutron diffusion equations by means of the finite difference method. For this purpose a realistic model of a sodium cooled fast reactor has been developed. We found that only for a discretization step width h equal to or less than the minimum diffusion length the results for the multiplication factor $k_{sub}(eff)$ and for the neutron flux reproduce the asymptotic values as h tends to zero (within practical accuracy requirements). Furthermore we have investigated the influence of the number of energy groups on $k_{sub}(eff)$ and found, for suitable chosen group boundaries, well decoupled behaviour of the effects of spatial and energetic resolution.

- 4639 ROLE OF THE CRBRP IN MEETING THE NATION'S ENERGY REQUIREMENTS. Riley, D.R. (Clinch River Breeder Reactor Plant Project, Oak Ridge, TN). pp 602-610 of Proceedings of the third annual UMR-MEC conference on energy: energy crisis, an evaluation of our resource potential. Morgan, J.D. (ed.). North Hollywood, CA; Western Periodicals Co. (1977).

From 3. annual conference on energy; Rolla, MO, USA (13 Oct 1976).

See CONF-761038--.

For over two decades the United States has been conducting research and development on the breeder reactor concept--an advanced reactor that breeds or produces more nuclear fuel than it consumes. The Clinch River Breeder Reactor Plant (CRBRP) is the nation's first large scale breeder designed to demonstrate the feasibility of producing electricity for commercial consumption. The current status of the CRBRP and its important role in meeting the nation's energy requirements are described.

- 4640 EXPERIMENTAL EVALUATION OF THE FLOW DISTRIBUTION IN THE CRBRP MODULAR STEAM GENERATOR. Lee, Y.J.; Weinberg, L.; Moriarty, M.P.; Oliva, R.M. New York; American Society of Mechanical Engineers (1977). 12p. (CONF-770912--3).

From ASME-IEEE joint power generation conference; Los Angeles, CA, USA (18 Sep 1977).

The shell side sodium flow velocity distribution was determined as a function of axial and circumferential position on a full scale model of the CRBRP Steam Generator, using water as the test fluid. Flow velocity profiles were obtained at 15 longitudinal locations, four of which provided data in three radial axes, to determine the length required to achieve a uniform flow distribution within the tube bundle and the flow distribution throughout the model. Velocity measurements in the inlet flow annulus indicated that the local flowrate at 180 degrees from the inlet was about 3 times the flowrate in line with the inlet. Velocity measurements in the outlet flow

annulus indicated relatively uniform flow conditions. Velocities in the lower stagnant regions were also measured and reported.

- 4641 ANALYSIS OF THE MECHANICAL RESPONSE OF LMFBR FUEL CLADS SUBJECTED TO IN-SERVICE PROPERTY VARIATIONS. Subbaraman, G. Blacksburg, VA; Virginia Polytechnic Inst. and State Univ. (1977). 125p. University Microfilms Order No. 77-22,089.

Thesis (Ph. D.).

Evidence from available literature is presented to illustrate the degradation of 316 type stainless steel considered for use in Liquid Metal Fast Breeder Reactors where sodium is used as the coolant. Nonuniform changes in properties of the steel have been found to occur due to the thermal, thermochemical, and irradiation environment to which it is exposed. Variations in impurity concentrations (such as carbon in the steel) of several orders of magnitude compared to the original values have been observed under controlled sodium exposures. At temperatures relevant to the reactor system migration of impurities by diffusion, compound formations and carburization/decarburization behavior have been observed to occur. Mechanical property measurements such as tensile and yield strengths made under these conditions indicate that thermal and thermochemical influences can result in variations in the above properties quite different from the original material. Modified formulations of the elastic and elastic-plastic analysis of the degraded fuel-clad are presented in two dimensions. The elastic and plastic parameters relating to the properties of the degraded material are represented by spatially varying functions as opposed to being treated as constants which is the conventional case. The changes in the mathematical nature of the constitutive equations are demonstrated by sample illustrations and solutions involving continuous changes in the elastic moduli through-the-thickness of the clad. Recommendations for the establishment of improved Reactor Research Development Standards are made based on the studies.

- 4642 RAPSDIE AND PHENIX PLANTS INTERMEDIATE HEAT EXCHANGERS EXPERIENCE. Goddett, J. (Centre de Marcoule, Bagnols, France); LaCroix, A.; Vial, B.; Sauvage, M. Trans. Am. Nucl. Soc.; 27: 221(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

PHENIX REACTOR;RAPSDIE REACTOR;HEAT EXCHANGERS;PERFORMANCE;REPAIR

- 4643 LIQUID-METAL VALVES. Markfort, D.; Hiltgen, H.; Kuehne, W.; Weber, H. (Postfach 5060, Bergisch, Germany). Trans. Am. Nucl. Soc.; 27: 222-224(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMFBR TYPE REACTORS;VALVES;DESIGN; REACTOR MATERIALS;PERFORMANCE;STRESSES; FABRICATION

- 4644 MATERIAL CONSIDERATION FOR LARGE SODIUM-HEATED STEAM GENERATORS. Duke, J.M.; Hebbard, M.; Reynolds, S.D.; Lewis, J. (Westinghouse Electric Corp., Tampa, FL). Trans. Am. Nucl. Soc.; 27: 224-225(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMFBR TYPE REACTORS;STEAM GENERATORS; REACTOR MATERIALS;MECHANICAL PROPERTIES; PERFORMANCE;DESIGN;CHROMIUM-MOLYBDENUM STEELS

- 4645 MATERIAL SELECTION FOR A SODIUM-HEATED STEAM GENERATOR. Grant, G.; Kakarala, C.R.; Hinton, B.M. (Babcock and Wilcox Co., Lynchburg, VA). Trans. Am. Nucl. Soc.; 27: 225-226(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMFBR TYPE REACTORS;STEAM GENERATORS; REACTOR MATERIALS;MECHANICAL PROPERTIES;DESIGN; CHROMIUM-MOLYBDENUM STEELS

- 4646 EFFECTS OF STOICHIOMETRY ON CLADDING ATTACK IN MIXED-OXIDE FUELS TO approx. ≈ 3.6 AT.% BURNUP. Hata, D.C.; Lawrence, L.A.; Weber, J.W. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 27: 233-235(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMFBR TYPE REACTORS;FUEL PINS;FUEL-CLADDING INTERACTIONS;PLUTONIUM DIOXIDE;URANIUM DIOXIDE;STAINLESS STEEL-316;PERFORMANCE TESTING; BURNUP;STOICHIOMETRY

- 4647 ANALYSIS OF FUEL-CLADDING CHEMICAL INTERACTION AT HIGH BURNUP. Lawrence, L.A.; Weber, J.W.; Devary, J.L. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 27: 235-236(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMFBR TYPE REACTORS;FUEL PINS;FUEL-CLADDING INTERACTIONS;BURNUP;PLUTONIUM DIOXIDE; URANIUM DIOXIDE;STAINLESS STEELS;PERFORMANCE TESTING

- 4648 IN-PILE HOMOGENIZATION OF MIXED-OXIDE FUEL. Mingst, B.; Olander, D.R. (Univ. of California, Berkeley). Trans. Am. Nucl. Soc.; 27: 236-237(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMFBR TYPE REACTORS;FUEL PINS;URANIUM DIOXIDE;PLUTONIUM DIOXIDE;BURNUP;PERFORMANCE TESTING;HOMOGENEOUS MIXTURES

- 4649 INTERACTION BETWEEN HYPERSTOICHIOMETRIC (U,PU)C AND MOLTEN STAINLESS-STEEL SHROUD MATERIAL. Latimer, T.W. (Los Alamos Scientific Lab., NM). Trans. Am. Nucl. Soc.; 27: 245-246(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMFBR TYPE REACTORS;FUEL PINS;FUEL-CLADDING INTERACTIONS;STAINLESS STEEL-316; PLUTONIUM CARBIDES;URANIUM CARBIDES;TEMPERATURE DEPENDENCE

- 4650 CALIBRATION SYSTEM FOR HYDROGEN METERS IN SODIUM. McKee, J.M. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 253(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMFBR TYPE REACTORS;STEAM GENERATORS; LEAKS;LEAK DETECTORS;HYDROGEN METERS; CALIBRATION;PERFORMANCE

- 4651 SODIUM-WATER REACTION TESTING IN THE LLTR. Neely, H.H.; DeBear, W.S.; Miller, A.E. (Atomics International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 27: 256-258(1977). From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977). See CONF-771109--.
- LMFBR. LMFBR TYPE REACTORS; STEAM GENERATORS; LEAKS; MOCKUP; TEST FACILITIES; TEMPERATURE GRADIENTS; PRESSURE GRADIENTS
- 4652 TH-U METALLIC FUEL: LMFBR POTENTIAL BASED ON EBR-II DRIVER-FUEL PERFORMANCE. Seidel, B.R.; Einziger, R.E.; Walter, C.M. (Argonne National Lab., Idaho Falls, ID). Trans. Am. Nucl. Soc.; 27: 282-283(1977). From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977). See CONF-771109--.
- LMFBR TYPE REACTORS; FUEL PINS; PERFORMANCE TESTING; URANIUM ALLOYS; THORIUM ALLOYS; SWELLING; FISSION PRODUCT RELEASE; FUEL-CLADDING INTERACTIONS
- 4653 FEASIBILITY OF IRRADIATING $^{233}\text{U}/^{232}\text{Th}$ METAL FUEL EXPERIMENTS IN EBR-II. Foltman, A.J.; Meneghetti, D. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 285-286(1977). From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977). See CONF-771109--.
- LMFBR. LMFBR TYPE REACTORS; FUEL PINS; THORIUM CYCLE; REACTOR KINETICS; REACTOR CORES; EBR-2 REACTOR
- 4654 STATUS OF MECHANICAL PROPERTIES AND MODELS FOR LMFBR AND LWR FUEL ELEMENT MODELING. Billone, M.C. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 298-299(1977). From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977). See CONF-771109--.
- LMFBR TYPE REACTORS; BWR TYPE REACTORS; PWR TYPE REACTORS; FUEL ELEMENTS; PERFORMANCE; HYDRAULICS; HEAT TRANSFER; URANIUM DIOXIDE; PLUTONIUM DIOXIDE; URANIUM CARBIDES; PLUTONIUM CARBIDES; URANIUM NITRIDES; PLUTONIUM NITRIDES; MATHEMATICAL MODELS; COMPUTER CODES
- 4655 REALITIES OF UTILIZING THORIUM-BASED FUELS IN LWRs AND LMFBRs. Loewenstein, W.B.; Sehgal, B.R. (Electric Power Research Inst., Palo Alto, CA). Trans. Am. Nucl. Soc.; 27: 312-313(1977). From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977). See CONF-771109--.
- LMFBR TYPE REACTORS; PWR TYPE REACTORS; BWR TYPE REACTORS; FUEL ELEMENTS; THORIUM CYCLE; REACTOR KINETICS; BREEDING RATIO; EFFICIENCY; PERFORMANCE
- 4656 LAMINAR MIXED-CONVECTION HEAT TRANSFER IN FINITE HEXAGONAL BUNDLES. Kim, J.Y.; Wolf, L. (Massachusetts Inst. of Tech., Cambridge). Trans. Am. Nucl. Soc.; 27: 384-385(1977). From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977). See CONF-771109--.
- LMFBR. LMFBR TYPE REACTORS; FUEL ASSEMBLIES; HYDRAULICS; LIQUID FLOW; TEMPERATURE GRADIENTS; HEAT TRANSFER
- 4657 CRITICALITY CALCULATIONS FOR FAST REACTOR FUEL STORAGE FACILITIES. Otter, J.M.; Pacora, V.J.; Specht, E.R. (Atomics International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 27: 412(1977). From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977). See CONF-771109--.
- LMFBR. LMFBR TYPE REACTORS; FUEL POOLS; SPENT FUEL STORAGE; CRITICALITY; NEUTRON TRANSPORT THEORY; REACTIVITY WORTHS
- 4658 DESIGN AND FABRICATION CONSIDERATIONS FOR CRBRP EX-VESSEL STORAGE TANK (EVST). Der Manuel, G. (Atomics International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 27: 485(1977). From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977). See CONF-771109--.
- CLINCH RIVER BREEDER REACTOR; SPENT FUEL STORAGE; SPECIFICATIONS
- 4659 PHYSICS DESIGN OF A SAFETY RESEARCH MODIFICATION OF EBR-II. Foltman, A.J.; Franklin, F.C.; Grimm, K.N.; Meneghetti, D. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 513-515(1977). From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977). See CONF-771109--.
- EBR-2 REACTOR; REACTOR EXPERIMENTAL FACILITIES; MODIFICATIONS; DESIGN; REACTOR CORES; REACTIVITY INSERTIONS
- 4660 GAS TAG DESIGNS FOR A GENERIC LMFBR. Figg, J.A. (Westinghouse Electric Corp., Richland, WA); Schenter, R.E.; DeClue, J.F. Trans. Am. Nucl. Soc.; 27: 578-579(1977). From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977). See CONF-771109--.
- CLINCH RIVER BREEDER REACTOR; FAILED ELEMENT MONITORS; FAILED ELEMENT DETECTION; FUEL ELEMENT FAILURE; PERFORMANCE
- 4661 EXPERIMENTS AND ANALYSES FOR AN LMFBR INHERENT SHUTDOWN SYSTEM. Specht, E.R.; Paschall, R.K.; Jackola, A.S.; Chung, D.K. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 581-582(1977). From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977). See CONF-771109--.
- LMFBR TYPE REACTORS; SCRAM; ENGINEERED SAFETY SYSTEMS; REACTOR PROTECTION SYSTEMS; SPECIFICATIONS
- 4662 MEASUREMENTS OF SODIUM OXIDE AGGLOMERATE PROPERTIES. Gieseke, J.A.; Reed, L.D. (Battelle Columbus Labs., OH). Trans. Am. Nucl. Soc.; 27: 616(1977). From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977). See CONF-771109--.
- LMFBR. LMFBR TYPE REACTORS; REACTOR ACCIDENTS; CONTAINMENT BUILDINGS; RADIOACTIVE AEROSOLS; SODIUM OXIDES; AGGLOMERATION
- 4663 SOME APPROXIMATIONS IN AEROSOL BEHAVIOR MODELS FOR SOURCE-TERM COMPUTATION. van de Vate, J.F. (Energy Research Foundation, Petten, Netherlands). Trans. Am. Nucl. Soc.; 27: 617-618(1977). From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977). See CONF-771109--.
- LMFBR. LMFBR TYPE REACTORS; REACTOR ACCIDENTS; CONTAINMENT BUILDINGS; RADIOACTIVE AEROSOLS

- 4664 GRAVITATIONAL AGGLOMERATION OF AEROSOLS. Pertmer, G.A.; Loyalka, S.K. (Univ. of Missouri, Columbia). Trans. Am. Nucl. Soc.; 27: 618-620(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION; CONTAINMENT BUILDINGS; RADIOACTIVE AEROSOLS; AGGLOMERATION
- 4665 APPLICATION OF ACOUSTIC AGGLOMERATORS FOR EMERGENCY USE IN LMFBR PLANTS. Wegrzyn, J.; Shaw, D.T.; Rajendran, N. (State Univ. of New York at Buffalo, Amherst). Trans. Am. Nucl. Soc.; 27: 620-621(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR ACCIDENTS; RADIOACTIVE AEROSOLS; AGGLOMERATION; CONTAINMENT SYSTEMS; AIR CLEANING SYSTEMS; PERFORMANCE
- 4666 COMPUTER MODEL FOR TRITIUM TRANSPORT IN AND RELEASE FROM THE CRBRP AND OTHER LMFBRs. Renner, T.A. (Argonne National Lab., NM); McPheeters, C.C.; McGuire, J.C. Trans. Am. Nucl. Soc.; 27: 621-622(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
CLINCH RIVER BREEDER REACTOR; LMFBR TYPE REACTORS; TRITIUM; RADIOACTIVITY TRANSPORT; PRIMARY COOLANT CIRCUITS; MATHEMATICAL MODELS
- 4667 MATERIAL RELEASE FOR VARIOUS FUELS FOLLOWING AN HCDA IN CRBR. Falotico, R.N.; Peddicord, K.L. (Oregon State Univ., Corvallis). Trans. Am. Nucl. Soc.; 27: 622-623(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
CLINCH RIVER BREEDER REACTOR; REACTOR CORE DISRUPTION; REACTOR CORES; FISSION PRODUCT RELEASE; PLUTONIUM DIOXIDE; URANIUM DIOXIDE; URANIUM CARBIDES; PLUTONIUM CARBIDES; URANIUM NITRIDES; PLUTONIUM NITRIDES; COMPARATIVE EVALUATIONS
- 4668 FUEL VAPOR TRANSPORT THROUGH A SODIUM POOL. Huang, T.C.; McDonald, J.W.; Still, R.K. (General Electric Co., Sunnyvale, CA). Trans. Am. Nucl. Soc.; 27: 623-625(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION; RADIOACTIVE AEROSOLS; COVER GAS; VAPORS
- 4669 CRBRP CONTAINMENT. Buhl, A.R. (Energy Research and Development Administration, Washington, DC); Piper, H.B. Trans. Am. Nucl. Soc.; 27: 646-647(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
CLINCH RIVER BREEDER REACTOR; CONTAINMENT BUILDINGS; CONTAINMENT SYSTEMS; PERFORMANCE; SPECIFICATIONS
- 4670 FIRST IN-REACTOR EXPERIMENT WITH SIMULATED LMFBR DEBRIS BED. Rivard, J.B. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 27: 653-654(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; MELTDOWN; CORIUM; RHR SYSTEMS; MOCKUP; SIMULATION; ACPR REACTOR; REACTOR EXPERIMENTAL FACILITIES
- 4671 EXPERIMENTS IN THE FORMATION OF PARTICULATE DEBRIS BEDS BY CONTACT OF MOLTEN CORE DEBRIS WITH SODIUM. Sowa, E.; Gabor, J.; Baker, L. Jr.; Pavlik, J.; Cassulo, J. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 654-655(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; CORIUM; SIMULATION; MELTDOWN; SODIUM; FUEL-COOLANT INTERACTIONS; MOCKUP
- 4672 ONSET OF THERMAL CONVECTION IN A FLUID-SATURATED POROUS LAYER WITH HEAT SOURCE. Hwang, I.T.; Marr, W.W. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 655-656(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; MELTDOWN; CORIUM; HEAT TRANSFER; RHR SYSTEMS; AFTER-HEAT REMOVAL
- 4673 FIRST IN-CORE MOLTEN FUEL POOL EXPERIMENT. Plein, H.G.; Carlson, G.A. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 27: 656-658(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; MELTDOWN; CORIUM; HEAT TRANSFER; MOCKUP; AFTER-HEAT REMOVAL; ACPR REACTOR; REACTOR EXPERIMENTAL FACILITIES
- 4674 INTERACTION OF HEAT-GENERATING MOLTEN URANIUM DIOXIDE WITH MAGNESIA BRICK. Stein, R.P.; Baker, L. Jr.; Gunther, W.H.; Cook, C. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 658-659(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; CORIUM; HEAT TRANSFER; AFTER-HEAT REMOVAL; RHR SYSTEMS; CORE CATCHERS; MAGNESIUM OXIDES; PERFORMANCE TESTING
- 4675 INTERACTIONS BETWEEN MOLTEN CORE DEBRIS AND CORE CONTAINMENT MATERIALS. Swanson, D.G. (Aerospace Corp., Washington, DC); Zehms, E.H.; Goddard, D.M.; Ang, C.Y.; van Paassen, H.L.; Kidwell, G.D.; Marchese, A.R. Trans. Am. Nucl. Soc.; 27: 659-660(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; MELTDOWN; CORIUM; HEAT TRANSFER; CORE CATCHERS; THERMAL STRESSES; THERMAL SHOCK; REACTOR MATERIALS; PERFORMANCE TESTING
- 4676 HEAT TRANSFER FROM A ROUND-BOTTOM POOL GENERATING HEAT INTERNALLY. Gabor, J.D.; Warner, J.G.; Cassulo, J.C.; Baker, L. Jr. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 660-662(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; MELTDOWN; CORIUM; AFTER-HEAT REMOVAL; HEAT TRANSFER; MOCKUP

- 4677 EFFECT OF THERMAL STRESS ON OXIDE CRUSTS. Ostensen, R.W. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 27: 662(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMEBR TYPE REACTORS; MELTDOWN; CORIUM; HEAT TRANSFER; LIQUID FLOW; THERMAL CONDUCTIVITY

- 4678 SOME COMMENTS ON SHOCK-INDUCED FRAGMENTATION AND DETONATING THERMAL EXPLOSIONS. Fauske, H.K. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 666-667(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMEBR TYPE REACTORS; FUEL ELEMENT FAILURE; FUEL-COOLANT INTERACTIONS; THERMAL SHOCK; FRAGMENTATION; FUEL PINS; TEMPERATURE GRADIENTS

- 4679 EFFECT OF INTERNAL RADIATION ON THE CONTACT INTERFACE TEMPERATURE BETWEEN UO_2 AND SODIUM. Cho, D.H. (Argonne National Lab., NM); Chan, S.H. Trans. Am. Nucl. Soc.; 27: 670-671(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMEBR TYPE REACTORS; FUEL-COOLANT INTERACTIONS; HEAT TRANSFER; TEMPERATURE GRADIENTS; THERMAL CONDUCTIVITY; SODIUM; URANIUM DIOXIDE

- 4680 THERMODYNAMIC PROPERTIES OF SODIUM FOR SAFETY ANALYSIS. Padilla, A. Jr. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 27: 671-672(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMEBR TYPE REACTORS; REACTOR ACCIDENTS; SODIUM; THERMODYNAMIC PROPERTIES

- 4681 USE OF GALLIUM AND ITS ALLOYS AS HIGH-TEMPERATURE COUPLANTS FOR ACOUSTIC TRANSDUCERS. Fitzgerald, J.W. (Woodmont Associates, Ltd., Bethesda, MD); Carey, W.M. Trans. Am. Nucl. Soc.; 27: 687-689(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMEBR TYPE REACTORS; REACTOR INSTRUMENTATION; TRANSDUCERS; SPECIFICATIONS

- 4682 INVESTIGATION OF THE IMPURITY DEPOSITION RATE OF THE PLUGGING INDICATOR. Yamamoto, H. (Atomic Energy Research Lab., Kawasaki, Japan); Izumi, S.; Doi, A.; Goto, T. Trans. Am. Nucl. Soc.; 27: 690-691(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMEBR TYPE REACTORS; PRIMARY COOLANT CIRCUITS; IMPURITIES; MONITORING; REACTOR INSTRUMENTATION; PERFORMANCE

- 4683 ACOUSTIC DETECTION OF BOILING IN SODIUM TEST LOOPS. Anderson, T.T.; Bobis, J.P.; Spencer, B.W.; Gavin, A.P. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 691-620(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMEBR TYPE REACTORS; PRIMARY COOLANT CIRCUITS; BOILING DETECTION; PERFORMANCE; ACOUSTIC

MONITORING

- 4684 PERFORMANCE OF FFTF-TYPE EDDY CURRENT FLOWMETERS IN THE SLSF P-SERIES TEST TRAINS. Pedersen, D.R.; Anderson, T.T.; Wilson, R.E. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 691-692(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR TYPE REACTORS; REACTOR INSTRUMENTATION; FLOWMETERS; PERFORMANCE TESTING

- 4685 SUPPORT FOR A NUCLEAR FUTURE: STUDENT RESPONSE TO THE CRBRP. Bremseth, M.D.; Clelland, D.A. (Univ. of Tennessee, Knoxville). Trans. Am. Nucl. Soc.; 27: 714-716(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

CLINCH RIVER BREEDER REACTOR; PUBLIC RELATIONS

- 4686 PLBR CAPITAL COST ASSESSMENT. Scarborough, J.C. (NUS Corp., Rockville, MD). Trans. Am. Nucl. Soc.; 27: 718-721(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

PLBR REACTOR; COST

- 4687 PROTOTYPE LARGE BREEDER REACTOR POOL CONCEPT. McDonald, J.S.; Brunings, J.E. (Atomics International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 27: 721-722(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

PLBR REACTOR; REACTOR CORES; DESIGN

- 4688 LARGE-AMPLITUDE THERMAL OSCILLATIONS MEASURED IN AN LMFBR STEAM GENERATOR TUBE. France, D.M.; Chiang, T.; Carlson, R.D. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 734-735(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR TYPE REACTORS; STEAM GENERATORS; HEAT TRANSFER; TEMPERATURE GRADIENTS; THERMAL STRESSES; LIQUID FLOW

- 4689 THERMAL-HYDRAULIC MODEL TESTS OF THE CRBR/IHTS THERMAL MIXER. Kasza, K.E.; Oras, J.J. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 735-737(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

CLINCH RIVER BREEDER REACTOR; HEAT EXCHANGERS; HYDRAULICS; HEAT TRANSFER; MIXING; LIQUID FLOW; TEMPERATURE DISTRIBUTION

- 4690 FLOW DISTRIBUTION IN EDGE SUBCHANNELS OF WIRE-WRAPPED 61-ROD ASSEMBLIES. Lin, E.I.H. (Argonne National Lab., IL); Schmitt, R.C.; Sha, W.T.; Goldner, F. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 741-743(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

See CONF-771109--.

LMFBR. LMEBR TYPE REACTORS; FUEL ASSEMBLIES; FUEL PINS; LIQUID FLOW; HYDRAULICS; HEAT TRANSFER; CONFIGURATION; FLOW RATE

4691 IMPACT OF BYPASS FLOW ON THERMAL TRANSIENTS IN LMFBR. Yang, J.W.; Agrawal, A.K. (Brookhaven National Lab., Upton, NY). Trans. Am. Nucl. Soc.; 27: 748-749(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR CORES; HYDRAULICS; LIQUID FLOW; HEAT TRANSFER; TEMPERATURE GRADIENTS

4692 ORIFIS: A COMPUTER CORE TO DISTRIBUTE REACTOR CORE COOLANT IN LARGE FBRS. McCurry, D.C.; Cantley, D.A. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 27: 749-750(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR CORES; HYDRAULICS; LIQUID FLOW; TEMPERATURE DISTRIBUTION; POWER DISTRIBUTION; COMPUTER CODES

4693 EFFECT OF NONUNIFORM GAPS ON COMPACTION OF AN LMFBR CORE. Kalinowski, J.E.; Swenson, D.V. (Westinghouse Advanced Reactors Division, Madison, PA). Trans. Am. Nucl. Soc.; 27: 750-751(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR CORES; FUEL ASSEMBLIES; DEFORMATION; SPACERS; STRESSES; REACTIVITY COEFFICIENTS

4694 DESIGN CONFIGURATIONS FOR CONTROLLING SODIUM VOID WORTH IN LARGE FBR CORES. Chandler, J.C.; Marr, D.R. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 27: 751-752(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR CORES; SODIUM; HYDRAULICS; LIQUID FLOW; VOID COEFFICIENT

4695 EFFECT ON PERFORMANCE OF CORE HEIGHT REDUCTION IN 1200-MW(E) LMFBRs. Su, S.F. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 752-753(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR CORES; SODIUM; HYDRAULICS; LIQUID FLOW; VOID COEFFICIENT; CONFIGURATION

4696 ASSESSMENT OF THE BREEDING POTENTIAL OF U--PU--ZR METAL-FUELED 1200-MW(E) LMFBRs. Lam, P.S.K.; Barthold, W.P. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 753-754(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR CORES; BREEDING RATIO; FUEL CYCLE; EFFICIENCY

4697 ADVANCED FUEL CORE DESIGNS FOR CRBRP. Galluzzo, N.G.; Vitti, J.A.; Brittingham, J.C.; von Arx, A.V.; Connors, P.A. (Atomic International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 27: 754-755(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
CLINCH RIVER BREEDER REACTOR; REACTOR CORES; SPECIFICATIONS; URANIUM CARBIDES; PLUTONIUM

CARBIDES; PERFORMANCE

4698 PIN-DIAMETER OPTIMIZATION OF HOMOGENEOUS AND TIGHTLY COUPLED HETEROGENEOUS ²³³U--TH METAL-FUELED LMFBRs. Lam, P.S.K.; Su, S.F. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 755-756(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; FUEL PINS; CONFIGURATION; OPTIMIZATION

4699 NUBOW-2D INELASTIC: A COMPUTER PROGRAM FOR THE BOWING HISTORY OF REACTOR CORES. Cha, B.K.; McLennan, G.A. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 756-757(1977).

From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR CORES; DEFORMATION; TWO-DIMENSIONAL CALCULATIONS; COMPUTER CODES

4700 DESIGN AND DEVELOPMENT OF THE CRBRP EX-VESSEL TRANSFER MACHINE. Jones, C.E. Jr. (Atomic International, Canoga Park, CA). Trans. Am. Nucl. Soc.; 27: 762-763(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
CLINCH RIVER BREEDER REACTOR; FUEL ASSEMBLIES; REMOTE HANDLING EQUIPMENT

4701 ANALYSIS OF STREAMING THROUGH THE PRIMARY COOLANT PIPE CHASEAWAY OF THE CLINCH RIVER BREEDER REACTOR. Williams, M.L.; Engle, W.W. Jr. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 27: 779-780(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
CLINCH RIVER BREEDER REACTOR; PRIMARY COOLANT CIRCUITS; RADIATION STREAMING; NEUTRONS; TWO-DIMENSIONAL CALCULATIONS; DISCRETE ORDINATE METHOD

4702 DEVELOPMENT AND TESTING OF A TRITIUM METER FOR SODIUM SYSTEMS. McPheeters, C.C. (Argonne National Lab., IL); Skladzien, S.B.; Osterhout, M.M. Trans. Am. Nucl. Soc.; 27: 786-787(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR COOLING SYSTEMS; SODIUM; TRITIUM; PROBES; MEASURING INSTRUMENTS

4703 IMPROVED DESIGN FOR AN ACTIVE CALORIMETER. Price, C.C. (Argonne National Lab., IL); English, P.N.; Courtney, J.C. Trans. Am. Nucl. Soc.; 27: 844-845(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
EBR-2 REACTOR; CALORIMETERS; DESIGN; LMFBR TYPE REACTORS; REACTOR CORES

4704 REDUCTION OF POWER DISTRIBUTION SENSITIVITY IN HETEROGENEOUS LMFBRs. Tzanos, C.P. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 900-901(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; POWER DISTRIBUTION;

 HETEROGENEOUS EFFECTS; REACTOR CORES; VOID
 COEFFICIENT

LMFBR TYPE REACTORS; THORIUM CYCLE; FUEL
 MANAGEMENT; PERFORMANCE; EFFICIENCY; REACTOR CORES

4705 PERFORMANCE OF 1000-MW(E) MIXED FUEL
 CYCLE LMFBRs. Shenoy, S.K.; Diaz, M.J.
 (Univ. of Florida, Gainesville). Trans. Am.
 Nucl. Soc.; 27: 901-902(1977).
 From ANS winter meeting; San Francisco, CA,
 USA (27 Nov 1977).
 See CONF-771109--.
²³³U--Th cycle. LMFBR TYPE REACTORS; VOID
 COEFFICIENT; REACTOR LATTICE PARAMETERS; FUEL
 CYCLE; THORIUM CYCLE

4706 EFFECT OF PLUTONIUM ISOTOPIC COMPOSITION
 ON LMFBR FUELING REQUIREMENTS. Jenquin, U.P.;
 Newman, D.F. (Battelle Northwest Labs.,
 Richland, WA). Trans. Am. Nucl. Soc.; 27:
 902-903(1977).
 From ANS winter meeting; San Francisco, CA,
 USA (27 Nov 1977).
 See CONF-771109--.
 LMFBR TYPE REACTORS; FUEL CYCLE; PLUTONIUM;
 ISOTOPE RATIO; REACTOR FUELING

4707 LMFBR REACTIVITY SURVEILLANCE USING EX-
 CORE DETECTORS. Lineberry, M.J.; Carpenter,
 S.G.; Beck, C.L. (Argonne National Lab.,
 Idaho Falls, ID). Trans. Am. Nucl. Soc.; 27:
 903-905(1977).
 From ANS winter meeting; San Francisco, CA,
 USA (27 Nov 1977).
 See CONF-771109--.
 LMFBR TYPE REACTORS; REACTIVITY METERS;
 COMPARATIVE EVALUATIONS; EFFICIENCY

4708 ²³³U--²³²Th FUEL CYCLE AS AN ALTERNATIVE
 TO LMFBRs. Karam, R.A. (Georgia Inst. of
 Tech., Atlanta). Trans. Am. Nucl. Soc.; 27:
 957(1977).
 From ANS winter meeting; San Francisco, CA,
 USA (27 Nov 1977).
 See CONF-771109--.
 LMFBR TYPE REACTORS; THORIUM CYCLE;
 FEASIBILITY STUDIES; BURNUP; FUEL MANAGEMENT

4709 PERFORMANCE CHARACTERISTICS OF THORIUM
 CYCLE IN CANDU REACTORS AND LMFBRs. Chang,
 Y.I.; Lam, P.S.K.; Till, C.E. (Argonne
 National Lab., IL). Trans. Am. Nucl. Soc.; 27:
 959-961(1977).
 From ANS winter meeting; San Francisco, CA,
 USA (27 Nov 1977).
 See CONF-771109--.
 LMFBR TYPE REACTORS; CANDU TYPE REACTORS;
 THORIUM CYCLE; COMPARATIVE EVALUATIONS;
 EFFICIENCY; BREEDING RATIO; FUEL MANAGEMENT

4710 USE OF THORIUM IN LMFBR BLANKETS.
 Driscoll, M.J. (Massachusetts Inst. of Tech.,
 Cambridge). Trans. Am. Nucl. Soc.; 27:
 962(1977).
 From ANS winter meeting; San Francisco, CA,
 USA (27 Nov 1977).
 See CONF-771109--.
 LMFBR TYPE REACTORS; BREEDING BLANKETS;
 THORIUM CYCLE; BREEDING RATIO; FUEL MANAGEMENT;
 EFFICIENCY

4711 PERFORMANCE OF VARIOUS FUEL CYCLES BASED
 ON THORIUM IN LMFBRs. Sehgal, B.R.; Naser, J.;
 Lin, C.; Loewenstein, W.B. (Electric Power
 Research Inst., Palo Alto, CA). Trans. Am.
 Nucl. Soc.; 27: 963(1977).
 From ANS winter meeting; San Francisco, CA,
 USA (27 Nov 1977).
 See CONF-771109--.

4712 FUEL HANDLING AND STORAGE SYSTEM FOR
 LARGE COMMERCIAL LMFBRs. Davis, C.R.;
 Sciacca, F.W. (General Electric Co.,
 Sunnyvale, CA). Trans. Am. Nucl. Soc.; 27:
 1020-1021(1977).
 From ANS winter meeting; San Francisco, CA,
 USA (27 Nov 1977).
 See CONF-771109--.
 LMFBR TYPE REACTORS; REACTOR MAINTENANCE;
 REMOTE HANDLING EQUIPMENT

4713 SURVEY OF ALTERNATIVE FUEL CYCLE OPTIONS
 WITHOUT PLUTONIUM SEPARATION. Till, C.E.;
 Chang, Y.I. (Argonne National Lab., IL).
 Trans. Am. Nucl. Soc.; 27: 428(1977).
 From ANS winter meeting; San Francisco, CA,
 USA (27 Nov 1977).
 See CONF-771109--.
 LMFBR TYPE REACTORS; PWR TYPE REACTORS; BWR
 TYPE REACTORS; CANDU TYPE REACTORS; FUEL CYCLE;
 REPROCESSING; REACTOR OPERATION; FEASIBILITY
 STUDIES

4714 LOW-DECONTAMINATION APPROACH TO A
 PROLIFERATION-RESISTANT FUEL CYCLE. Asquith,
 J.G.; Grantham, L.F. (Atomics International,
 Washington, DC). Trans. Am. Nucl. Soc.; 27:
 433-434(1977).
 From ANS winter meeting; San Francisco, CA,
 USA (27 Nov 1977).
 See CONF-771109--.
 LMFBR; PWR; BWR. FUEL CYCLE; REPROCESSING;
 LMFBR TYPE REACTORS; BWR TYPE REACTORS; PWR TYPE
 REACTORS; FEASIBILITY STUDIES; REACTOR KINETICS

4715 DENATURED FAST REACTOR SYMBIOSIS.
 Burns, T.J. (Oak Ridge National Lab., TN);
 White, J.R.; Bartine, D.E.; Kallfelz, J.M.;
 Rowland, D.M. Trans. Am. Nucl. Soc.; 27: 454-
 456(1977).
 From ANS winter meeting; San Francisco, CA,
 USA (27 Nov 1977).
 See CONF-771109--.
 LMFBR. LMFBR TYPE REACTORS; FUEL CYCLE;
 DENATURED FUEL; PLANNING; URANIUM 233; URANIUM 238

4716 LARGE PLANT THORIUM-FUELED REACTOR
 SYSTEM CHARACTERISTICS. Felten, L.D.;
 Galluzzo, N.G.; Brittingham, J.C. (Atomics
 International, Canoga Park, CA). Trans. Am.
 Nucl. Soc.; 27: 456-457(1977).
 From ANS winter meeting; San Francisco, CA,
 USA (27 Nov 1977).
 See CONF-771109--.
 PLBR REACTOR; FUEL CYCLE; URANIUM CARBIDES;
 THORIUM CARBIDES; REACTOR KINETICS

4717 SENSITIVITY ANALYSIS FOR FAST REACTOR
 FUEL CYCLE COST. Badruzzaman, A.; Becker, M.
 (Rensselaer Polytechnic Inst., Troy, NY).
 Trans. Am. Nucl. Soc.; 27: 457-458(1977).
 From ANS winter meeting; San Francisco, CA,
 USA (27 Nov 1977).
 See CONF-771109--.
 LMFBR. LMFBR TYPE REACTORS; FUEL CYCLE;
 ECONOMICS; COST; EFFICIENCY

4718 SURFACE OXIDATION PROCESS PRIOR TO
 IGNITION OF A SODIUM DROPLET. Tsai, S.S.
 (Brookhaven National Lab., Upton, NY). Trans.
 Am. Nucl. Soc.; 27: 524-525(1977).
 From ANS winter meeting; San Francisco, CA,
 USA (27 Nov 1977).
 See CONF-771109--.

- LMFBR TYPE REACTORS;FIRES;SODIUM;OXIDATION;
DROPLETS
- 4719 FLUX SYNTHESIS PROGRAMME KASY FOR THE APPROXIMATE SOLUTION OF THE THREE-DIMENSIONAL MULTIGROUP-NEUTRON-DIFFUSION-EQUATION. PRESENT STATE, PLANNED IMPROVEMENTS AND APPLICATIONS TO REACTOR DESIGN PROBLEMS. Buckel, G. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik); de Wouters, R.; Pilate, S. (Societe Belge pour l'Industrie Nucleaire, Brussels). Atomkernenergie; 30: No. 2, 82-87(1977).
From Technical meeting on coarse-mesh methods: progress in the methods and their application to reactor problems; Erlangen, Germany, F.R. (25 - 26 Jan 1977).
This paper describes the three-dimensional flux synthesis programme KASY and its application to practical reactor problems. The recent extensions or improvements of KASY comprise: a) solution of the adjoint problem, allowing three-dimensional perturbation calculations in connection with the three-dimensional evaluation programme AUDI 3; b) connection to the space-dependent reactor dynamics programme RAUMZEIT; c) inclusion into the modular programme system KAPROS, allowing the execution of three-dimensional reactor analysis by combined use of KASY and finite difference programmes. A compilation of the main applications of KASY to fast reactor analysis performed from 1970 up to now is presented. It contains in particular: a) evaluation of experiments performed in critical facilities as a support of SNR-300 design (ksub(eff), control rod worths, power mapping); b) application of KASY to the routine SNR-300 (and SNR-2) calculations in three-dimensional hexagonal geometry, for which the method proved efficient in spite of the case complexity; burn-up behaviour was taken into account.
- 4720 UPGRADING THE EBR-II REACTOR SHUTDOWN SYSTEM. Gale, N.L. (Argonne National Lab., Idaho Falls, ID). Trans. Am. Nucl. Soc.; 27: 845-846(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
EBR-2 REACTOR; REACTOR CONTROL SYSTEMS; REACTOR SHUTDOWN; MODIFICATIONS; DESIGN
- 4721 CLADDING BREACHES IN MIXED-OXIDE FUEL PINS. Weber, J.W.; Almassy, M.Y. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 27: 227-228(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
FFTF REACTOR; FUEL CANS; FUEL ELEMENT FAILURE; STAINLESS STEEL-304; STAINLESS STEEL-316; PERFORMANCE TESTING; PLUTONIUM DIOXIDE; URANIUM DIOXIDE
- 4722 PHENOMENOLOGICAL MODEL FOR MIXED-OXIDE FUEL PIN CLADDING BREACHES. Almassy, M.Y. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 27: 228-229(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
FFTF REACTOR; FUEL CANS; FUEL ELEMENT FAILURE; STAINLESS STEEL-316; PLUTONIUM DIOXIDE; URANIUM DIOXIDE; PERFORMANCE TESTING
- 4723 EFFECTS OF LOCAL CESIUM CONCENTRATIONS ON MIXED-OXIDE FUEL BEHAVIOR. Karnesky, R.A. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 27: 229-231(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
FFTF REACTOR; FUEL PINS; PLUTONIUM DIOXIDE; URANIUM DIOXIDE; CESIUM; VOIDS; PERFORMANCE
- 4724 PERFORMANCE OF COMMERCIALY PRODUCED MIXED-OXIDE FUELS IN EBR-II. Hales, J.W.; Hoth, C.W. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 27: 237-238(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
FFTF REACTOR; FUEL PINS; PERFORMANCE TESTING; PLUTONIUM DIOXIDE; URANIUM DIOXIDE; BURNUP; PHYSICAL RADIATION EFFECTS; STAINLESS STEEL-316
- 4725 PERFORMANCE OF COMMERCIALY PRODUCED FULL-SIZE FFTF FUEL PINS IN GETR. Hales, J.W.; Baker, R.B.; Fuller, R.E. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 27: 238-239(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
FFTF REACTOR; FUEL PINS; PERFORMANCE TESTING; PLUTONIUM DIOXIDE; URANIUM DIOXIDE; STAINLESS STEEL-316
- 4726 EFFECTS OF POWER AND IRRADIATION BEHAVIOR ON EBR-II BLANKET MANAGEMENT. Koenig, J.F. (Argonne National Lab., Idaho Falls, ID); Chang, L.K.; Cutforth, D.C.; Feldman, E.E.; Meneghetti, D.; Rempert, R.H.; Seidel, B.R. Trans. Am. Nucl. Soc.; 27: 248-249(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
EBR-2 REACTOR; FUEL ASSEMBLIES; BREEDING BLANKETS; SWELLING; PERFORMANCE; BURNUP; FUEL MANAGEMENT
- 4727 OPERATIONAL EXPERIENCE WITH EBR-II WATER-TO-SODIUM LEAK DETECTION SYSTEM. Osterhout, M.M. (Argonne National Lab., Idaho Falls, ID). Trans. Am. Nucl. Soc.; 27: 253-255(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
EBR-2 REACTOR; STEAM GENERATORS; LEAKS; LEAK DETECTORS; HYDROGEN METERS; PERFORMANCE
- 4728 FFTF DYNAMIC MODELING AND SENSITIVITY ANALYSIS. Kerlin, T.W.; Huynh, Q.A. (Univ. of Tennessee, Knoxville). Trans. Am. Nucl. Soc.; 27: 362-364(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
FFTF REACTOR; REACTOR KINETICS; REACTOR STABILITY
- 4729 FUEL MOTION IN TREAT TRANSIENT-OVER-POWER TEST E6. Stanford, G.S.; Doerner, R.C.; DeVolpi, A.; Stewart, R.R.; Regis, J.P. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 496(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
FFTF REACTOR; FUEL PINS; FUEL ELEMENT FAILURE; TESTING; TREAT REACTOR; TRANSIENTS

- 4730 MOBILE HELIUM COOLING LOOP SYSTEM FOR IN-REACTOR PAHR EXPERIMENTS. Brammer, J.A. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 27: 651-653(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109---.
LMFBR. LMFBR TYPE REACTORS;AFTER-HEAT REMOVAL;RHR SYSTEMS;MOCKUP;PERFORMANCE TESTING; ACPR REACTOR;REACTOR EXPERIMENTAL FACILITIES
- 4731 STRUCTURAL ANALYSES OF THE EBR-II PRIMARY SYSTEM PIPING FOR THE EBR-II SRM OPERATING CONDITIONS. Chang, L.K.; Wu, T.S.; Feldman, E.E. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 740-741(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109---.
EBR-2 REACTOR;PRIMARY COOLANT CIRCUITS; MODIFICATIONS;HYDRAULICS;LIQUID FLOW; TEMPERATURE GRADIENTS
- 4732 BORE-SIDE INSPECTION OF EBR-II STEAM GENERATOR TUBES. Whitham, G.K.; Longua, K.J. (Argonne National Lab., Idaho Falls, ID). Trans. Am. Nucl. Soc.; 27: 759-760(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109---.
EBR-2 REACTOR;STEAM GENERATORS;TUBES; INSPECTION;REMOTE CONTROL;ULTRASONIC TESTING
- 4733 EXPERIENCE WITH BREACHED ELEMENTS AT EBR-II. Lambert, J.D.B. (Argonne National Lab., IL); Kirn, F.S.; Ebersole, E.R. Trans. Am. Nucl. Soc.; 27: 820-822(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109---.
EBR-2 REACTOR;FUEL ELEMENT FAILURE;FAILED ELEMENT DETECTION;PRIMARY COOLANT CIRCUITS; FISSION PRODUCTS
- 4734 FISSION PRODUCT CONTROL AT EBR-II. Monson, L.R. (Argonne National Lab., Idaho Falls, ID); McDaniel, M.J.; Knox, A.E.; Rice, R.E.; Olson, W.H.; Ruther, W.E. Trans. Am. Nucl. Soc.; 27: 822-823(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109---.
EBR-2 REACTOR;FUEL ELEMENT FAILURE;FISSION PRODUCT RELEASE;PRIMARY COOLANT CIRCUITS
- 4735 PREPARATIONS AT EBR-II FOR BREACHED ELEMENT OPERATION. Sackett, J.I.; Fryer, R.M.; Buschman, H.W.; Rice, R.E.; Barak, W.S. (Argonne National Lab., Idaho Falls, ID). Trans. Am. Nucl. Soc.; 27: 823-825(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109---.
EBR-2 REACTOR;FUEL ELEMENT FAILURE;REACTOR SAFETY;FISSION PRODUCT RELEASE
- 4736 DESIGN AND USE OF THE FISSION PRODUCT SOURCE AT EBR-II. Strain, R.V.; Fogle, G.L.; Thresh, H.R.; Heinrich, R.R. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 825-826(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109---.
EBR-2 REACTOR;IRRADIATION CAPSULES;FISSION PRODUCTS;RADIATION SOURCES;RECOILS
- 4737 FISSION GAS BEHAVIOR IN THE EBR-II PRIMARY SYSTEM. So, B.Y.C. (Argonne National Lab., IL); Armstrong, J.R.; Ebersole, E.R. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 826-827(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109---.
EBR-2 REACTOR;FISSION PRODUCT RELEASE;FAILED ELEMENT DETECTION;COVER GAS
- 4738 ANALYSIS OF THE TRANSIENT CORRECTION FACTOR IN TREAT REACTOR EXPERIMENTS. Graff, D.L. (Argonne National Lab., IL); Hungerford, H.E. Trans. Am. Nucl. Soc.; 27: 947-948(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109---.
TREAT REACTOR;LMFBR TYPE REACTORS;FUEL PINS; POWER COEFFICIENT;CONFIGURATION
- 4739 FFTF CORE COMPONENT MEASURING SYSTEM. Mullen, R.E. (Aerojet Manufacturing Co., Fullerton, CA); Martin, R.L. Trans. Am. Nucl. Soc.; 27: 981-982(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109---.
FFTF REACTOR;REACTOR CORES;REACTOR COMPONENTS;REMOTE HANDLING EQUIPMENT; CONFIGURATION
- 4740 FFTF ELEMENT MEASURING SYSTEM. Mullen, R.E. (Aerojet Manufacturing Co., Fullerton, CA); Harbison, S.G.; Martin, R.L. Trans. Am. Nucl. Soc.; 27: 982-984(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109---.
FFTF REACTOR;REACTOR CORES;FUEL PINS;CONTROL ELEMENTS;REMOTE HANDLING EQUIPMENT; CONFIGURATION
- 4741 SAFETY OF FAST REACTORS. Teague, H.J. pp 183-209 of Nuclear reactor safety. Farmer, F.R. (ed.). New York; Academic Press, Inc. (1977).
The following topics are briefly discussed: special features important to LMFBR safety, reactor kinetics, loss of flow, fuel handling hazards, reactor core disruption, and containment. (DG)
- 4742 HEAT CONDUCTION BY MWR USING SEMI-INFINITE MEDIA TRIAL FUNCTIONS. Weber, D.P.; Gettings, M.E. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 388-389(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109---.
LMFBR. LMFBR TYPE REACTORS;REACTOR CORE DISRUPTION;REACTOR SAFETY;HYDRAULICS;HEAT TRANSFER;TEMPERATURE GRADIENTS;MATHEMATICAL MODELS
- 4743 EMPIRICAL FUEL MOTION MODEL FOR TREAT LOSS-OF-FLOW TEST F1. Palm, R.G.; Gehl, S.M.; Fink, C.L.; Dever, D.J.; Rothman, A.B. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 497(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109---.
LMFBR. LMFBR TYPE REACTORS;LOSS OF FLOW; MELTDOWN;SIMULATION;TREAT REACTOR;URANIUM DIOXIDE;MATHEMATICAL MODELS;MELTING

- See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; LOSS OF FLOW;
SIMULATION; HYDRODYNAMICS; FUEL ELEMENT FAILURE
- 4744 FRAGMENTATION DYNAMICS OF MOLTEN UO_2 IN SODIUM UNDER TOP CONDITIONS. Jones, B.G. (Univ. of Illinois, Urbana); Spencer, B.W.; Henry, R.E. Trans. Am. Nucl. Soc.; 27: 497-498(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; EXCURSIONS; FUEL ELEMENT FAILURE; FUEL-COOLANT INTERACTIONS; FRAGMENTATION; URANIUM; SODIUM; MELTDOWN; HYDRODYNAMICS
- 4745 28-PIN CLADDING RELOCATION EXPERIMENTS. Henry, R.E.; Jeans, W.C.; Quinn, D.J.; Spleha, E.A. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 498-500(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; LOSS OF FLOW; MELTDOWN; SIMULATION; HYDRODYNAMICS
- 4746 INVESTIGATION OF MOLTEN-FUEL RELOCATION DYNAMICS WITH SIMULANT FLUIDS. Chun, M.H.; Ginsberg, T.; Jones, O.C. Jr. (Brookhaven National Lab., Upton, NY). Trans. Am. Nucl. Soc.; 27: 500-501(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; MELTDOWN; SIMULATION; HYDRODYNAMICS
- 4747 FREEZING OF A FLOWING LIQUID INSIDE A MELTING TUBE. Yim, A. (Argonne National Lab., IL); Epstein, M.; Bankoff, S.G.; Lambert, G.A. Trans. Am. Nucl. Soc.; 27: 501-502(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; MELTDOWN; SIMULATION; FREEZING
- 4748 PLATEOUT AND THERMAL LOADING CONSIDERATIONS OF FUEL EJECTED DURING TRANSIENT- OVER-POWER ACCIDENTS IN LMFBRs. Wong, K.W.; Dhir, V.K.; Kastenberger, W.E. (Univ. of California, Los Angeles). Trans. Am. Nucl. Soc.; 27: 502-503(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; TRANSIENTS; EXCURSIONS; HYDRODYNAMICS; FUEL-COOLANT INTERACTIONS
- 4749 CAMEL SINGLE-PIN FUEL SWEEPOUT TEST C2. Spencer, B.W.; Bova, L.; Goldfuss, G.T.; Raridon, D.; Henry, R.E.; Armstrong, D.R. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 503-505(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; EXCURSIONS; FUEL-COOLANT INTERACTIONS; SODIUM; URANIUM DIOXIDE; HYDRODYNAMICS; SIMULATION
- 4750 POST-TEST EVALUATION OF THE SLF LOSS-OF-FLOW SAFETY EXPERIMENT P2. Bordner, G.L.; Gartside, C.H.; Hwang, I.T.; Marr, W.W.; McDonald, T.A.; Tessier, J.H.; Thompson, D.H.; Dunn, F.E. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 506-507(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
- 4751 POST-TEST ANALYSIS OF TRANSIENT OVERPOWER TREAT TEST R9 FOR BOL FFTF-TYPE FUEL. Semenza, L.A.; Koopman, R.N. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 507-508(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
FTTF REACTOR; REACTIVITY INSERTIONS; TRANSIENTS; EXCURSIONS; SIMULATION; FUEL ELEMENT FAILURE
- 4752 EOS SERIES OF EXPERIMENTS IN TREAT. Tylka, J.P.; Rothman, A.B.; Froehle, P.H.; Pokorny, G.J.; Stephenson, M.E.; Ulrich, A.J. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 508(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; MELTDOWN; SIMULATION; FUEL ELEMENT FAILURE; TREAT REACTOR; HYDRODYNAMICS
- 4753 HYDRODYNAMIC ANALYSIS OF THE LMFBR PROMPT BURST EXCURSION (PBE) EXPERIMENT. Young, M.F. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 27: 508-510(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; EXCURSIONS; SIMULATION; HYDRODYNAMICS; DATA; TIME DEPENDENCE
- 4754 NONPROTOTYPIC EFFECTS DUE TO FUEL LENGTH AND NUMBER OF PINS IN IN-PILE TESTING. Nguyen, D.H. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 27: 511-512(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION; SIMULATION; MOCKUP; FUEL PINS; LENGTH
- 4755 CODED APERTURE IMAGING OF REACTOR-ILLUMINATED LMFBR FUEL PINS. Kelly, J.G.; Stalker, K.T. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 27: 512-513(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION; TEST FACILITIES; SIMULATION; REMOTE VIEWING EQUIPMENT; NEUTRON CAMERAS; IMAGES; FUEL PINS
- 4756 STOCHASTIC PROCESSES IN THERMAL-HYDRAULIC FLOW FIELDS. Hurt, R. (Bradley Univ., Peoria, IL); Doerner, R.C.; Meek, C.C. Trans. Am. Nucl. Soc.; 27: 515-516(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; FUEL ELEMENT FAILURE; STOCHASTIC PROCESSES; MATHEMATICAL MODELS; HEAT TRANSFER
- 4757 ABOVE-CORE STRUCTURAL RESPONSE TO A MOLTEN CORE DISASSEMBLY. Ash, J.E. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 517-518(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).

- See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION; STRESSES; MECHANICAL STRUCTURES; REACTOR INTERNALS; DEFORMATION
- 4758 ANALYSIS OF SLUG IMPACT AND COOLANT SPILLAGE USING A HYBRID LAGRANGIAN--EULERIAN METHOD. Chu, H.Y.; Chang, Y.W.; Fistedis, S.H. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 518-520(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR VESSELS; STRESSES; REACTOR CORE DISRUPTION; IMPACT SHOCK; CLOSURES; SODIUM; DEFORMATION; HYDRODYNAMICS; MATHEMATICAL MODELS; STRESS ANALYSIS
- 4759 COMPARISONS OF REXCO AND ICEPEL CODE CALCULATIONS WITH TESTS OF PLASTICALLY DEFORMABLE PIPES FILLED WITH COMPRESSIBLE FLUIDS. Nagumo, G. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 520-522(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR ACCIDENTS; CONTAINMENT; CONTAINMENT SYSTEMS; STRESSES; PIPES; DEFORMATION; COMPUTER CALCULATIONS; COMPARATIVE EVALUATIONS
- 4760 COMPARISON OF COMPUTER CODE PREDICTIONS WITH COVA EXPERIMENTS. Chang, Y.W.; Guildys, J.; Fistedis, S.H. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 522-524(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR ACCIDENTS; CONTAINMENT; CONTAINMENT SYSTEMS; REACTOR VESSELS; STRESSES
- 4761 STATISTICAL PARAMETRIC ANALYSES OF SODIUM FIRE. Person, L.W.; Vaurio, J.K. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 525-526(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; FIRES; SODIUM; STATISTICS; MATHEMATICAL MODELS
- 4762 FILTRATION OF SODIUM FIRE AEROSOLS. Jordan, S.; Alexas, A.; Lindner, W. (Gesellschaft fuer Kernforschung mbH, Karlsruhe, Germany). Trans. Am. Nucl. Soc.; 27: 526-527(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; FIRES; RADIOACTIVE AEROSOLS; FILTRATION; SODIUM; AIR FILTERS; PERFORMANCE
- 4763 LARGE-SCALE SODIUM INTERACTIONS: PRELIMINARY TEST RESULTS FOR LIMESTONE CONCRETE. Smaardyk, J.E.; Sutherland, H.J.; King, D.L.; Dahlgren, D.A. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 27: 528(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION; CHEMICAL REACTION KINETICS; CONCRETES; SODIUM; SIMULATION; DECOMPOSITION; CORROSIVE EFFECTS
- 4764 THERMAL BEHAVIOR OF CONCRETE. Knight, R.L. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 27: 528-529(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR MATERIALS; CONCRETES; DECOMPOSITION; SODIUM; CHEMICAL REACTION KINETICS; REACTOR CORE DISRUPTION; CORROSIVE EFFECTS
- 4765 EVALUATION OF THE EFFECT OF SHUTDOWN CRACKS IN TREAT-TESTED FUEL PINS. Scott, J.H. (Los Alamos Scientific Lab., NM). Trans. Am. Nucl. Soc.; 27: 530-531(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; FUEL PINS; FUEL PELLETS; CRACKS; PHYSICAL RADIATION EFFECTS; TREAT REACTOR; TRANSIENTS; SIMULATION; REACTOR CORE DISRUPTION; FUEL ELEMENT FAILURE; SIMULATION
- 4766 APPLICATION OF LARSON--MILLER PARAMETER TO LMFBR FUEL FAILURE DYNAMICS. Sun, Y.H.; Henry, T.P. (Brookhaven National Lab., Upton, NY). Trans. Am. Nucl. Soc.; 27: 531-532(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; FUEL ELEMENT FAILURE; MATHEMATICAL MODELS; TRANSIENTS; CALCULATION METHODS
- 4767 COMBINED MOTION OF FUEL AND COOLANT UNDER HIGH-RAMP TRANSIENT OVERPOWER CONDITIONS. Chang, K.I.; Cho, D.H. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 532-533(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; EXCURSIONS; TRANSIENTS; MELTDOWN; HYDRODYNAMICS; FUEL-COOLANT INTERACTIONS; MATHEMATICAL MODELS; COMPUTER CALCULATIONS
- 4768 PLUTO2: A COMPUTER CODE FOR THE ANALYSIS OF OVERPOWER ACCIDENTS IN LMFBRs. Wider, H.U. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 533-534(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; EXCURSIONS; TRANSIENTS; FUEL ELEMENT FAILURE; MATHEMATICAL MODELS; HYDRODYNAMICS; COMPUTER CODES; P CODES
- 4769 SIMPLIFIED METHOD OF COMPUTING CLADDING AND FUEL STRAIN DURING A SLOW OVERPOWER TRANSIENT. Frank, M.V.; Kastenber, W.E. (Univ. of California, Berkeley). Trans. Am. Nucl. Soc.; 27: 534-536(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; FUEL CANS; FUEL ELEMENTS; TRANSIENTS; STRESSES; STRAINS; FUEL-CLADDING INTERACTIONS; COMPUTER CODES; MATHEMATICAL MODELS
- 4770 FAILURE PATTERN WITHIN AN FTR SUBASSEMBLY UNDER A TOP ACCIDENT. Yung, S.C.; Wilburn, N.P. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 27:

- 536-537(1977).
From ANS winter meeting; San Francisco, CA,
USA (27 Nov 1977).
See CONF-771109---.
FFTF REACTOR;FUEL ASSEMBLIES;TRANSIENTS;FUEL
ELEMENT FAILURE;MATHEMATICAL MODELS
- 4771 BOILING INCOHERENCY EFFECTS IN
UNPROTECTED LOSS-OF-FLOW ACCIDENTS.
Theofanous, T.G. (Purdue Univ., West
Lafayette, IN); Miao, C.; Lois, L. Trans. Am.
Nucl. Soc.; 27: 536-537(1977).
From ANS winter meeting; San Francisco, CA,
USA (27 Nov 1977).
See CONF-771109---.
LMFBR. LMFBR TYPE REACTORS;REACTOR SAFETY;
LOSS OF FLOW;FUEL ASSEMBLIES;HYDRAULICS;BOILING;
HEAT TRANSFER
- 4772 INTERSUBASSEMBLY INCOHERENCIES AND
GROUPING TECHNIQUES IN LMFBR HYPOTHETICAL
OVERPOWER ACCIDENT. Wilburn, N.P.
(Westinghouse Electric Corp., Richland, WA).
Trans. Am. Nucl. Soc.; 27: 538-539(1977).
From ANS winter meeting; San Francisco, CA,
USA (27 Nov 1977).
See CONF-771109---.
LMFBR TYPE REACTORS;REACTOR SAFETY;REACTOR
ACCIDENTS;FUEL ELEMENT FAILURE;POWER
DISTRIBUTION
- 4773 FUSS: THE INTEGRATED FUMO, SAS3A,
NEUTRONICS UPDATE SYSTEM. Martin, F.J.
(Westinghouse Electric Corp., Richland, WA).
Trans. Am. Nucl. Soc.; 27: 539-541(1977).
From ANS winter meeting; San Francisco, CA,
USA (27 Nov 1977).
See CONF-771109---.
FFTF REACTOR;REACTOR SAFETY;LOSS OF FLOW;
MATHEMATICAL MODELS;REACTOR KINETICS;COMPUTER
CODES;F CODES
- 4774 NATURAL-CIRCULATION DECAY HEAT REMOVAL
IN AN LMFBR. Perkins, K.R.; Bari, R.A.;
Albright, D.C. (Brookhaven National Lab.,
Upton, NY). Trans. Am. Nucl. Soc.; 27: 542-
543(1977).
From ANS winter meeting; San Francisco, CA,
USA (27 Nov 1977).
See CONF-771109---.
LMFBR TYPE REACTORS;LOSS OF FLOW;HYDRAULICS;
HEAT TRANSFER;NATURAL CONVECTION;COMPUTER CODES;
MATHEMATICAL MODELS;AFTER-HEAT REMOVAL
- 4775 CURL: A TRANSIENT SIMULATION CODE FOR
LMFBRs. Cady, K.B. (Cornell Univ., Ithaca,
NY); Pavlenko, G.F.; Raber, J.R. Khatib-Rahbar,
M. (Cornell Univ., Ithaca, NY); Pavlenko,
G.F.; Raber, J.R. Trans. Am. Nucl. Soc.; 27:
543-545(1977).
From ANS winter meeting; San Francisco, CA,
USA (27 Nov 1977).
See CONF-771109---.
LMFBR TYPE REACTORS;REACTOR SAFETY;
TRANSIENTS;REACTOR KINETICS;HEAT TRANSFER;
HYDRAULICS;C CODES;COMPUTER CODES
- 4776 PLOOP CODE FOR THE PRIMARY LOOP COOLANT
FLOW AND HEAT TRANSFER ANALYSIS OF CERTAIN POOL-
TYPE LMFBRs. Kalimullah. (Argonne National
Lab., IL). Trans. Am. Nucl. Soc.; 27: 545-
546(1977).
From ANS winter meeting; San Francisco, CA,
USA (27 Nov 1977).
See CONF-771109---.
LMFBR TYPE REACTORS;REACTOR SAFETY;
TRANSIENTS;HYDRAULICS;HEAT TRANSFER;P CODES;
COMPUTER CODES
- 4777 EFFECTS OF NONCONDENSIBLE VOID FRACTION
ON FREEZING OF FLOWING FLUIDS. Greene, G.A.
(Brookhaven National Lab., Upton, NY); Jones,
O.C. Jr.; Kazimi, M.S. Trans. Am. Nucl. Soc.;
27: 546-547(1977).
From ANS winter meeting; San Francisco, CA,
USA (27 Nov 1977).
See CONF-771109---.
LMFBR. LMFBR TYPE REACTORS;REACTOR SAFETY;
MELTDOWN;CORIUM;SOLIDIFICATION;TWO-PHASE FLOW
- 4778 FLOW REGIME OBSERVATIONS IN TWO-PHASE
VOLUME-HEATED BOILING POOLS. Ginsberg, T.;
Jones, O.C. Jr. (Brookhaven National Lab.,
Upton, NY). Trans. Am. Nucl. Soc.; 27: 547-
549(1977).
From ANS winter meeting; San Francisco, CA,
USA (27 Nov 1977).
See CONF-771109---.
LMFBR. LMFBR TYPE REACTORS;REACTOR CORE
DISRUPTION;MELTDOWN;REACTOR SAFETY;TWO-PHASE
FLOW;LIQUID FLOW;BOILING;CORIUM
- 4779 FOAM: PRINCIPAL FLOW REGIME IN A VOLUME-
HEATED BOILING POOL. Epstein, M.; Condiff,
D.W. (Argonne National Lab., IL). Trans.
Am. Nucl. Soc.; 27: 549-550(1977).
From ANS winter meeting; San Francisco, CA,
USA (27 Nov 1977).
See CONF-771109---.
LMFBR. LMFBR TYPE REACTORS;REACTOR SAFETY;
MELTDOWN;TWO-PHASE FLOW;LIQUID FLOW;BOILING;F
CODES;COMPUTER CODES
- 4780 SIMPLIFIED MULTICOMPONENT PHASE
TRANSITION MODEL. Suo-Anttila, A.J. (Los
Alamos Scientific Lab., NM). Trans. Am. Nucl.
Soc.; 27: 550-551(1977).
From ANS winter meeting; San Francisco, CA,
USA (27 Nov 1977).
See CONF-771109---.
LMFBR. LMFBR TYPE REACTORS;REACTOR SAFETY;
REACTOR CORE DISRUPTION;MELTDOWN;LIQUID FLOW;
HEAT TRANSFER;MATHEMATICAL MODELS;TRANSITION
FLOW
- 4781 EFFECTS OF INTERNAL CIRCULATION VELOCITY
AND NONCONDENSIBLE GAS ON VAPOR CONDENSATION
FROM A RISING BUBBLE. Ozisik, M.N. (North
Carolina State Univ., Raleigh, NC); Kress, T.S.
Trans. Am. Nucl. Soc.; 27: 551-552(1977).
From ANS winter meeting; San Francisco, CA,
USA (27 Nov 1977).
See CONF-771109---.
LMFBR. LMFBR TYPE REACTORS;REACTOR SAFETY;
REACTOR CORE DISRUPTION;FUEL-COOLANT
INTERACTIONS;GASES;BUBBLES;VAPOR CONDENSATION;
NATURAL CONVECTION;HEAT TRANSFER
- 4782 SIMMER-VENUS II COMPARISON FOR
MECHANICAL DISASSEMBLY OF AN LMFBR.
Henninger, R.J.; Alcouffe, R.E. (Los Alamos
Scientific Lab., NM). Trans. Am. Nucl. Soc.;
27: 553-554(1977).
From ANS winter meeting; San Francisco, CA,
USA (27 Nov 1977).
See CONF-771109---.
LMFBR TYPE REACTORS;REACTOR SAFETY;REACTOR
CORE DISRUPTION;POWER DISTRIBUTION;HEAT
TRANSFER;REACTIVITY COEFFICIENTS;S CODES;V
CODES;COMPUTER CODES;COMPARATIVE EVALUATIONS
- 4783 SYSTEM DAMAGE POTENTIAL FOR CRBRP.
Hakim, S.J. (Argonne National Lab., IL);
Abramson, P.B.; Henninger, R.J. Trans. Am.
Nucl. Soc.; 27: 554-555(1977).

- From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
CLINCH RIVER BREEDER REACTOR; REACTOR SAFETY; REACTOR CORE DISRUPTION; HYDRAULICS; REACTOR INTERNALS; DAMAGE; DEFORMATION
- 4784 SIMMER-I ACCIDENT CONSEQUENCE CALCULATIONS. Bell, C.R.; Boudreau, J.E. (Los Alamos Scientific Lab., NM). Trans. Am. Nucl. Soc.; 27: 555-556(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
CLINCH RIVER BREEDER REACTOR; REACTOR CORE DISRUPTION; REACTOR SAFETY; REACTOR CORES; FUEL ASSEMBLIES; DEFORMATION; DAMAGE; S CODES
- 4785 STOCHASTICALLY VARIABLE PARAMETERS IN PROBABILISTIC ACCIDENT ANALYSIS. Abramson, P.B.; Hummel, H.H.; Gelbard, E.M.; Pizzica, P.A.; Sienicki, J.J. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 556-558(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR CORE DISRUPTION; HEAT TRANSFER; HYDRAULICS; REACTIVITY COEFFICIENTS; MATHEMATICAL MODELS
- 4786 TRANSPORT/DIFFUSION COMPARISONS FOR SMALL-CORE LMFBR DISRUPTIVE ACCIDENTS. Tomlinson, E.T. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 27: 558-559(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR CORE DISRUPTION; REACTIVITY COEFFICIENTS; NEUTRON DIFFUSION EQUATION; PERTURBATION THEORY; TRANSPORT THEORY; COMPARATIVE EVALUATIONS
- 4787 ENERGY INCREASE FROM BUBBLE COLLAPSE FOLLOWING RECRITICALITY IN LARGE MOLTEN POOLS. Fuller, E.L. (Electric Power Research Inst., Palo Alto, CA). Trans. Am. Nucl. Soc.; 27: 559-560(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; MELTDOWN; CORIUM; CRITICALITY; BUBBLES; REACTIVITY COEFFICIENTS
- 4788 VOID REACTIVITY EFFECTS ON FAST REACTOR HCDA EXCURSION ENERGY. Munno, F.J.; Sheaks, D.J.; Rhodes, C.J.; Shimizu, M. (Univ. of Maryland, College Park). Trans. Am. Nucl. Soc.; 27: 559-560(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR CORE DISRUPTION; MELTDOWN; REACTIVITY COEFFICIENTS; MATHEMATICAL MODELS; EXCURSIONS; REACTOR SAFETY
- 4789 BUBBLE COLLAPSE REACTIVITY INCREASES IN BOILING FUEL/STEEL POOLS. Abramson, P.B.; Daly, T.A.; Lell, R.M.; Gelbard, E.M. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 562-563(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY;
- REACTOR CORE DISRUPTION; MELTDOWN; CORIUM; BUBBLES; VOID COEFFICIENT; MATHEMATICAL MODELS
- 4790 MAXIMUM WAKE TEMPERATURE AND NUSSELT NUMBER BEHIND BLOCKAGES IN SODIUM-COOLED BUNDLES. Han, J.T.; Fontana, M.H.; Wantland, J.L. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 27: 544-545(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; FUEL ASSEMBLIES; FLOW BLOCKAGE; TEMPERATURE DISTRIBUTION; HYDRAULICS; NUSSELT NUMBER; REACTOR SAFETY
- 4791 MODEL TO CORRELATED RESIDENCE TIME MEASUREMENTS BEHIND BLOCKAGES. Han, J.T. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 27: 565-566(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; FUEL ASSEMBLIES; FUEL PINS; FLOW BLOCKAGE; HYDRAULICS; LIQUID FLOW; REACTOR SAFETY
- 4792 CALCULATED VS EXPERIMENTAL TEMPERATURE DISTRIBUTION BEHIND A SIX-CHANNEL CENTRAL BLOCKAGE. Dearing, J.F. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 27: 565-566(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; FLOW BLOCKAGE; FUEL ASSEMBLIES; FUEL PINS; TEMPERATURE DISTRIBUTION; MOCKUP; MATHEMATICAL MODELS; COMPARATIVE EVALUATIONS
- 4793 DYNAMIC BOILING TESTS IN A 19-PIN SIMULATED LMFBR FUEL ASSEMBLY. Wantland, J.L.; Clapp, N.E.; Fontana, M.H.; Gnadt, P.A.; Hanus, N. (Oak Ridge National Lab., TN). Trans. Am. Nucl. Soc.; 27: 567-569(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR ACCIDENTS; FUEL ASSEMBLIES; FUEL PINS; PRESSURE DROP; LIQUID FLOW; HEAT TRANSFER; BOILING; TEMPERATURE GRADIENTS
- 4794 APPROACH FOR TREATING TWO-DIMENSIONAL COOLANT BOILING IN AN LMFBR FUEL SUBASSEMBLY. Marr, W.W.; Hwang, I.T. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 569-570(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; FUEL ASSEMBLIES; REACTOR SAFETY; LOSS OF FLOW; LIQUID FLOW; HYDRAULICS; BOILING; MATHEMATICAL MODELS; TWO-DIMENSIONAL CALCULATIONS
- 4795 SOME COMMENTS ON SODIUM BEHAVIOR (BOILING) IN LMFBR WIRE-WRAPPED SUBASSEMBLIES. Fauske, H.K. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 570-571(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR SAFETY; LOSS OF FLOW; FUEL ASSEMBLIES; FUEL PINS; HYDRAULICS; LIQUID FLOW; BOILING; TWO-PHASE FLOW; MATHEMATICAL MODELS

- 4796 DETECTION OF FUEL ASSEMBLY FLOW REDUCTIONS IN A HETEROGENEOUS CORE. Howard, P.A. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 571-572(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; FUEL ASSEMBLIES; FLOW BLOCKAGE; HYDRAULICS; LIQUID FLOW; FLOW RATE; FUEL ELEMENT FAILURE; TEMPERATURE GRADIENTS
- 4797 ACOUSTIC ANALYSIS FROM SODIUM BOILING TESTS USING THORS BUNDLE 6A. Sheen, S.H.; Bobis, J.P.; Carey, W.M. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 572-573(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR ACCIDENTS; FUEL ASSEMBLIES; FUEL PINS; HYDRAULICS; LIQUID FLOW; FLOW RATE; MOCUP; BOILING; TEMPERATURE GRADIENTS; TWO-PHASE FLOW
- 4798 RATE EFFECTS IN THE DYNAMIC VAPORIZATION OF URANIUM DIOXIDE. Benson, D.A. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 27: 574-575(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; FUEL PINS; MELTDOWN; URANIUM DIOXIDE; EVAPORATION; TEMPERATURE DEPENDENCE; PRESSURE DEPENDENCE
- 4799 THEORETICAL CONSIDERATIONS OF RATE EFFECTS IN URANIUM DIOXIDE. Gorham-Bergeron, E. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 27: 575-576(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; FUEL PINS; MELTDOWN; URANIUM DIOXIDE; EVAPORATION; TEMPERATURE DEPENDENCE; PRESSURE DEPENDENCE
- 4800 EFFECTIVE EQUATION-OF-STATE MEASUREMENTS ON URANIUM DIOXIDE. Reil, K.O. (Sandia Labs., Albuquerque, NM); Cronenberg, A.W. Trans. Am. Nucl. Soc.; 27: 576-577(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; MELTDOWN; FUEL PINS; URANIUM DIOXIDE; VAPOR PRESSURE; TEMPERATURE DEPENDENCE; PRESSURE DEPENDENCE; EQUATIONS OF STATE
- 4801 DISPERSIVE POTENTIAL OF IRRADIATED BREEDER REACTOR FUEL DURING A THERMAL TRANSIENT. Gruber, E.E.; Deitrich, L.W. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 577-578(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; REACTOR SAFETY; LOSS OF FLOW; REACTOR CORE DISRUPTION; FUEL ELEMENT FAILURE; PRESSURE GRADIENTS; GASES
- 4802 COMPARISON OF SAS3D LOF TRANSIENT ANALYSES FOR 1200-MW(E) LMFBR CONVENTIONAL AND NONCONVENTIONAL CORE DESIGNS. Shih, T.A.; Tenne, M.I. (General Electric Co., Sunnyvale, CA). Trans. Am. Nucl. Soc.; 27: 578-579(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR CORE DISRUPTION; PRESSURE GRADIENTS; GASES
- 4803 PIN PLENUM FISSION GAS RELEASE MODEL TO DESCRIBE FISSION GAS EFFECTS IN A VOIDED CHANNEL. Bowers, C.H.; Spencer, B.W. (Argonne National Lab., IL). Trans. Am. Nucl. Soc.; 27: 582-583(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR. LMFBR TYPE REACTORS; FUEL PINS; REACTOR SAFETY; FUEL ELEMENT FAILURE; LOSS OF FLOW; GASES; PRESSURE GRADIENTS; MATHEMATICAL MODELS
- 4804 FISSION GAS BUBBLE MODELING FOR LMFBR ACCIDENTS. Ostensen, R.W. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 27: 583-584(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; REACTOR SAFETY; REACTOR CORE DISRUPTION; PRESSURE GRADIENTS; GASES
- 4805 HYDRAULIC RESPONSE OF AN LMFBR PRIMARY SYSTEM TO PIPE RUPTURE. Madni, I.K.; Agrawal, A.K.; Cazzoli, E.G.; Guppy, J.G. (Brookhaven National Lab., Upton, NY). Trans. Am. Nucl. Soc.; 27: 739-740(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS; LOSS OF COOLANT; PRIMARY COOLANT CIRCUITS; HYDRAULICS; FLOW RATE; LIQUID FLOW; PRESSURE GRADIENTS
- 4806 DIMENSIONAL INSTABILITY OF CF8 STAINLESS-STEEL CASTINGS AT ELEVATED TEMPERATURES. Hull, F.C. (Westinghouse Research and Development Center, Pittsburgh); Eiffler, E.E. Trans. Am. Nucl. Soc.; 27: 221-222(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
STAINLESS STEELS; CASTINGS; HIGH TEMPERATURE; VERY HIGH TEMPERATURE; SHRINKAGE; CRYSTAL-PHASE TRANSFORMATIONS; PHASE DIAGRAMS; AGING; HEAT TREATMENTS; FFTF REACTOR
- 4807 DESIGN FEATURES AND OBJECTIVES OF THE FUELS AND MATERIALS EXAMINATION FACILITY. Doman, D.R.; Trego, A.L.; Burgess, C.A. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 27: 984-985(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
HOT CELLS; DESIGN; SPENT FUELS; REMOTE HANDLING EQUIPMENT; FFTF REACTOR
- 4808 DESIGN AND DEVELOPMENT OF THE EXAMINATION EQUIPMENT FOR THE FUELS AND MATERIALS EXAMINATION FACILITY. Trego, A.L.; Frick, D.C.; Weber, E.T. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 27: 985(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
FFTF REACTOR; HOT CELLS; DESIGN; REMOTE HANDLING EQUIPMENT; SPENT FUELS
- 4809 ADVANTAGES OF EQUIPMENT TESTING IN IEM

- CELL MOCKUP. LeGore, T.; Dietrick, D.L.; Engstrom, J.W.; Richman, R.B. (Westinghouse Electric Corp., Richland, WA). Trans. Am. Nucl. Soc.; 27: 987-988(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
HOT CELLS;REMOTE HANDLING EQUIPMENT;FFTF REACTOR;SPENT FUELS;DESIGN
- 4810 SPENT FUEL SHIPPING CASK DESIGNS FOR BREEDER REACTORS. Allen, G.C. Jr.; Dupree, S.A.; Eakes, R.G.; Freedman, J.M.; Pope, R.B. (Sandia Labs., Albuquerque, NM). Trans. Am. Nucl. Soc.; 27: 780-781(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
CLINCH RIVER BREEDER REACTOR;FFTF REACTOR; SPENT FUEL CASKS;DESIGN
- 4811 SPENT FUEL SHIPPING SYSTEM FOR LARGE COMMERCIAL LMFBRs. Davis, C.R. (General Electric Co., Sunnyvale, CA). Trans. Am. Nucl. Soc.; 27: 1021-1022(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
SPENT FUEL CASKS;DESIGN;LMFBR TYPE REACTORS; SAFETY;SPENT FUELS
- 4812 EFFECT OF INTERNAL THERMAL RADIATION ON THE CONTACT INTERFACE TEMPERATURE. Cho, D.H. (Argonne National Lab., IL); Chan, S.H. Lett. Heat Mass Transfer; 4: No. 6, 465-475(1977).
The transient energy transfer following a sudden contact between two media initially at different temperatures was investigated, taking into account the effect of internal radiation in the hot medium. Analytical expressions for the contact interface temperature valid for small times are derived. Illustrative calculations are presented for a sudden contact between molten uranium dioxide and sodium.
- 4813 INTRODUCTION TO MATERIALS CONSIDERATION FOR LMFBR COMPONENTS. Morabito, J.J.; Purdy, C.M. (Energy Research and Development Administration, Washington, DC). Trans. Am. Nucl. Soc.; 27: 220-221(1977).
From ANS winter meeting; San Francisco, CA, USA (27 Nov 1977).
See CONF-771109--.
LMFBR TYPE REACTORS;REACTOR MATERIALS; MECHANICAL PROPERTIES;WELDED JOINTS;FABRICATION; CORROSION
- 4814 EXPERIMENTAL STUDY OF THE CONSEQUENCES OF ACCIDENTAL SODIUM RELEASE IN THE FORM OF PEROXIDE AEROSOLS ON THE TOMATO AND THE LETTUCE. Camus, H.; Delmas, J.; Grauby, A.; Disdier, R. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Service d'Etudes et Recherches sur l'Environnement). Bull. Inf. Sci. Tech. (Paris); No. 222, 61-70(1977). (In French).
Industrial development of sodium cooled breeder reactors have lead to the study of consequences of accidents occurring in the cooling circuit. The effects on crops when aerosols are released in the atmosphere can be determined after a certain amount of molten sodium (stable element ^{23}Na) is brought into contact with the open air. Experiments are performed in a 'Phytotron' equipment on lettuces and tomatoes in dry atmosphere at two different moments during the growing process. The aerosol studied has a high content of sodium peroxide and the ground deposit varied between 17 and 450 mg/m² sodium equivalent. (Concentration at distances of 800 m and 1500 m calculated for a theoretical fire involving 10 tons of sodium). Necrosis, visible only with a microscope, were reported when deposits in sodium equivalent were under 300 mg/m². Leaves were destroyed if necrosis were numerous (deposit above 300 mg/m²). Tomatoes, and fruits in particular, were found to be more resistant than lettuces. Sodium embedded in dead cells does not migrate in the plant and does not disturb the plants physiological equilibrium. New sprouts have normal sodium percentage. The consequences are essentially burns of which the extent is more or less high depending on the deposits and the kind of species involved.
- 4815 STUDY OF FORCED CONVECTION BOILING OF SODIUM AT LOW HEAT FLUXES IN ELECTRICALLY HEATED SEVEN-ROD BUNDLES. Eisenhaver, S.W. Seattle; Univ. of Washington (1977). 165p. University Microfilms Order No. 77-26,820.
Thesis (Ph. D.).
An experimental study of forced convection boiling of sodium in electrically heated seven-rod bundles of fast reactor subchannel geometry is made. These conditions are similar to those found in in-core gamma-heated sodium boiling experiments. Analysis of the experimental data shows that various boiling patterns can be generated, depending upon the degree of incipient superheat, the amount of inlet throttling and the energy storage potential of the system. The voiding patterns associated with these parameters are related to the observed experimental signals and explained in terms of the controlling physical processes. The behavior of pressure signals generated during boiling is found to be highly time dependent and is accounted for by variations in the noise source and the attenuation properties of an expanding two-phase region. Calculations with the SAS3A computer code (10) show good qualitative agreement with the experiments. For slow transients in small bundles at low power, the multi-bubble model used in the code is a suitable approximation. Discrepancies between the code predictions and experimental data are explained in terms of the computational model. The importance of adequately accounting for system heat capacity and external heat losses is demonstrated. The design and analysis of in-core, gamma-heated boiling generators are examined in the light of these results. Significantly different voiding scenarios can be produced by variations in bundle design or operation. The expected neutronic and acoustic signals for several idealized voiding patterns are considered.
- 4816 MECHANISTIC STUDY OF FUEL FREEZING, CHANNEL PLUGGING, AND CONTINUED COOLABILITY DURING FAST REACTOR OVERPOWER EXCURSIONS. Wong, K.W. Los Angeles; Univ. of California (1977). 184p. University Microfilms Order No. 77-25,355.
Thesis (Ph. D.).
The work presents a mechanistic model which describes events following fuel pin failure which may lead to in-channel fuel plate-out. The thermal and hydraulic effects of the plate-out fuel are also evaluated. Results of the model indicate that during a slow transient power accident, in-core fuel freezing and channel plugging may be expected to occur. The extent of fuel freezing and channel plugging is sensitive to fuel particle size, amount of fuel injected and amount of fission gas injected into the coolant channel. Limiting curves for the initiation of melting can be established to indicate conditions for melting of wire wrap and adjoining cladding due to plate-out fuel.

Results predicted by this model are found to be consistent with the TREAT H-4 experiment, when the particle size and amount of fuel injected are adjusted to match the inlet flow rate.

- 4817 ADVANCED LMFBR FUELS. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977). 705p. (CONF-771008--).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

Separate abstracts are included for forty-two of the forty-six papers presented concerning the national advanced LMFBR fuels development program; irradiation testing experience for oxide fuels; recent developments in properties of advanced fuels; irradiation testing experience for carbide and nitride fuels; safety aspects of advanced fuels; and irradiation performance analysis. Four papers have been previously included in the data base.

- 4818 US ADVANCED LMFBR FUELS DEVELOPMENT PROGRAM. Simmons, J.M.; Leary, J.A.; Kittel, J.H.; Cox, C.M. pp 2-14 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Following the oil crisis in 1974, a national Advanced LMFBR Fuels Development Program was initiated in the U. S. This program was developed on the basis of the experience obtained during the exploratory years. As a result, most aspects of advanced fuels development have been expanded in the U. S. in a unified national program. The experience obtained during the exploratory phase has been summarized previously. The purpose of this paper is to describe the new program and to summarize recent major findings.

- 4819 UK STRATEGY AND PROGRAMMES FOR ADVANCED LMFBR FUELS. Bishop, J.F.W.; Chamberlain, A.; Gatley, J.A. pp 15-25 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

The first task for the UK is to develop a reliable plutonium oxide fuelled LMFBR with the associated plutonium storage, fabrication, transport, reprocessing and waste disposal parts of the complete fuel cycle in time for a substantial installation programme in AD 2000. This means a full scale demonstration in the first half of the 1990's. The second task is the establishment of this developed reactor system as part of the electricity generation network. For the purpose of this paper, two timescales can be delineated for objectives. The first is for oxide LMFBR systems. It is directed to a Demonstration Point of 1990 to 1995 and a continuing schedule leading to what we may call Fully Developed Oxide. The second is for other fuels and coolants where no dates have been fixed for Commercial Availability, but where these are definitely known to be later than the oxide timescale.

- 4820 SNR-CARBIDE PROGRAM: ITS POSITION BETWEEN PRESENT AND FUTURE. Karsten, G. (Gesellschaft fuer Kernforschung, Karlsruhe, Ger.). pp 26-50 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Information on the development of carbide-fueled LMFBR reactors is presented concerning ecological perspectives; the nuclear fuel cycle; fast reactor strategy; and program review and fuel pin concept.

- 4821 FRENCH ADVANCED FUELS DEVELOPMENT PROGRAM. Pascard, R. (Centre d'Etudes Nucleaires de Fontenay-aux-Roses, France). pp 52-60 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

The aspects of the French carbide program are described. Since the beginning, it has experienced some inflexions but its main guidelines are maintained. In particular, the Na-bonded fuel element is still the reference concept to be developed in priority. With maybe the exception of reprocessing, where obviously some effort is needed, the actual program, as it is shaped, should bring sufficient knowledge to make possible in a few years a full assessment of the future of carbide fuel in fast reactors.

- 4822 JAPANESE DEVELOPMENT PROGRAM FOR ADVANCED LMFBR FUELS. Nomura, S. (Japan Atomic Energy Research Inst., Tokai); Shimokawa, J.; Uematsu, K.; Noro, K. pp 61-73 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Information on the Japanese LMFBR fuels program is presented concerning the necessity and development program for advanced fuels; facility for mixed carbide, carbonitride, and nitride fuels; irradiation facilities in JOYO MK-2 core; and development program of advanced oxide fuels.

- 4823 DEVELOPMENT OF ADVANCED FUELS IN SWITZERLAND. Stratton, R.W. (Swiss Federal Inst. for Reactor Research, Wuerenlingen). pp 74-85 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

The development program for advanced carbide fuels for LMFBR reactors is described.

- 4824 CNEN PROGRAM ON ADVANCED LMFBR FUELS. Cecchi, P.; Covarelli, M.; Marucci, G.; Nobili, A. pp 86-93 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

CNEN activity on fast reactor fuels has been directed, in the past years, to the following purposes: study of basic phenomena affecting the in-pile behavior of fuel pins and feasibility studies of the vented-to-coolant fuel element and vibrocompacted fuel.

- 4825 UK IRRADIATION EXPERIENCE RELEVANT TO ADVANCED OXIDE FUEL CONCEPTS FOR LMFBR'S. Swanson, K.M.; Sloss, W.; Batey, W. pp 95-109

of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Most of the irradiation of oxide FR fuel in the UK has been in support of the driver charge fuel for PFR. Advanced oxide fuels mean fuels appreciably greater in linear rating and diameter than PFR driver charge fuels. The basic parameters of the two irradiated sub-assembly experiments to be described are given, along with those of the PFR driver charge for contrast. The irradiations were done in DFR, which completed its final run in March of this year, and the post irradiation examinations were done in the Dounreay cells.

4826 EXPERIENCE AND FUTURE PLAN OF FAST REACTOR OXIDE FUELS IRRADIATION IN JAPAN.

Uematsu, K. (Power Reactor and Nuclear Fuel Development Corp., Tokyo) Ishida, Y.; Komatsu, J. pp 110-121 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

The irradiation of mixed oxide fuels in Japan has been directed so far to the development of the JOYO MK-I core fuels. This program has been successfully completed and proved the satisfactory behavior and stability of JOYO type fuels. Current irradiation program is mostly directed to MONJU type fuel pins and JOYO MK-II fuel pins. JOYO MK-II core will be used for the irradiation testing facility for advanced fuel and materials. Besides the steady-state irradiation experiments, non-steady-state irradiation experiments will be emphasized in the future. They are overpowered experiments, run to failure experiments, which are to define the performance limit, and load followed experiments. Defective fuel pin irradiation experiments are also programmed in the loop of JOYO MK-II core. Future program and some results obtained so far for fuel pins, subassembly, and cladding materials are described.

4827 MECHANICAL BEHAVIOR OF A FUEL PIN FOR FAST REACTORS: EFFECT OF THE FUEL FORM AND OF THE APPROACH-TO-POWER ON THE CHANGE OF THE FUEL STACK LENGTH UNDER IRRADIATION.

Calza-Bini, A.; Caracchini, R.; Cosoli, G.; Filacchioni, G.; Lanchi, M.; Nobili, A.; Pesce, E.; Rocca, U.V.; Rotoloni, P.L. pp 122-135 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Within the framework of an experimental programme, several fuel pins were irradiated. Measurements have been made on the change of the length of the fuel stack and of the cladding with power, temperature, and burnup. These experiments were done as an overall check of the pin behavior as well as to provide a key to interpret the contribution of the different phenomena to the observed changes. The present work consists of the analysis of the results obtained during three experiments in which the fuel was kept at a high heat rating. The behavior of solid and annular pellets has been compared; the effect of different approach-to-power rates has also been taken into account.

ADVANCED OXIDE FUELS. Cox, C.M. (Hanford Engineering Development Lab., Richland, WA); Hilbert, R.F.; Biancheria, A. pp 136-148 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

In June, 1977, 210 advanced oxide fuel elements had been irradiated or were under irradiation in EBR-II. By December, 1977, this total will be 294. Included are such design parameters as larger pin diameters, smaller cladding thicknesses, and higher fuel smear densities than the reference FFTF fuel element design. Structural materials include 316 and D9 stainless steels, Inconel 706, and PE 16.

4829 ADVANCED MIXED OXIDE FUEL ELEMENT FOR FAST REACTORS.

Delpyroux, P.; Arnaud, G.; Ollier, H.; Mogniot, J. (CEA-CEN, Cadarache, France). pp 149-152 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

After definitive specification of the mixed oxide fuel element for Superphenix 1, two avenues were open in France for the development of the next generation of fuel element for this reactor type: a carbide fuel element which had been under lengthy investigation by C.E.A.; and an advanced mixed oxide fuel element adapted to the heterogeneous-core concept. The basic concept being pursued in the preliminary element project for Superphenix 2 is outlined here.

4830 PROPERTIES AND IRRADIATION BEHAVIOR OF ADVANCED OXIDE PARTICLE FUEL.

Lahr, H.W.H. pp 153-164 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Compared with the conventional pellet concept, fabrication of particle fuel has advantages in economy and in the reduction of the radiation burden to the personnel due to its capability of being automated. Moreover, the process is a very variable one; with minor modifications, oxides and carbides can be prepared for light water reactors, fast breeders and high-temperature reactors. Numerous irradiation tests did not yield disadvantages in the operating behavior, which are specifically attributable to this fuel variant. The tests described, which were, above all, intended to check the startup behavior of particle columns, prove that certain apprehensions concerning this fuel are unjustified for this phase of the operation.

4831 MIXED-NITRIDE FUEL PERFORMANCE IN EBR-II.

Bauer, A.A. (Battelle Columbus Labs., OH); Cybulskis, P.; Green, J.L. pp 299-312 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Data have been obtained for a total of 42 fuel pins containing mixed-nitride fuel irradiated in EBR-II. Thirty-eight of the pins contained sodium-bonded fuel and four helium-bonded fuel. No failures occurred in the helium-bonded pins or in sodium-bonded pins in which

the fuel was supported by internal shrouds. Fission-gas release data are presented along with observations concerning fuel and fuel-pin dimensional and structural changes and the influence of design and operating parameters on pin performance.

- 4832 UK IRRADIATION EXPERIENCE RELEVANT TO ADVANCED CARBIDE FUEL CONCEPTS FOR LMFBR'S. Bagley, K.Q.; Batey, W.; Paris, R.; Sloss, W.M.; Snape, G.P. pp 313-325 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Despite discouraging prognoses of fabrication and reprocessing problems, it is recognized that the quest for a carbide fuel pin design which fully exploits the favourable density and thermal conductivity of (U,Pu) monocarbide must be maintained. Studies in aid of carbide fuel development have, therefore, continued in the UK in parallel with those on oxide, albeit at a substantially lower level of effort, and a sufficient body of irradiation experience has been accumulated to allow discrimination of realistic fuel pin designs.

- 4833 GERMAN CARBIDE IRRADIATION TEST EXPERIENCE FOR FAST REACTORS. Kummerer, K.R. (Kernforschungszentrum, Karlsruhe, Ger.). pp 326-342 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Most of the open problems in carbide pin design and performance are of a very simple nature. As was the case with mixed oxide one or two decades ago, we must get experience in the parameter field of design and operation conditions. The main design parameters are carbide composition, density, internal geometry. The operation conditions in question are temperature and linear rod power and, of course, the achievable burnup. This field of experience should include steady-state as well as power-cycling and transient conditions. Finally the performance of representative carbide bundle structures are to be considered.

- 4834 FRENCH IRRADIATION TEST EXPERIENCE FOR CARBIDE FUELS. Combette, P. (Centre d'Etudes Nucleaires de Fontenay-aux-Roses, France). pp 343-347 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

A research program has been performed at Commissariat a l'Energie Atomique to develop a carbide fuel element, for use in French breeder reactors. The major portion of the program is the evaluation of the performance of fuel elements irradiated in Osiris (thermal flux) or in Rapsodie (fast flux) reactors. The main parameters under investigation are the fuel pellet density (79 to 97 percent T.D., which corresponds to a smear density of 71 to 86 percent), the linear heat rating (70 to 140 kW/m), the burn up (up to 120,000 MWD/t) and the type of bond (He or Na). Up to now, a single grade of cladding material has been tested, solution annealed 316 stainless steel.

CARBIDE FUEL. Stratton, R.W.; Smith, L. (Swiss Federal Inst. for Reactor Research, Wuerenlingen). pp 348-360 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

A major part of the EIR programme developing (UPu)C sphere-pac fuel is the irradiation testing and subsequent post irradiation examination (P.I.E.) of vibro-filled fuel pins. Initial operating conditions were set to give maximum clad temperatures of approximately 650°C (sometimes exceeded) and peak linear ratings of 100 kW/m (often not attained). The FILOS series of irradiations in the EIR reactor SAPHIR have mean burn-up targets of 1 percent fima, most of the others in DIDO (UK), BR-2 (Belgium) and D.F.R. (Scotland) have reached mean values between 4 and 6.4 percent fima. All the tests reported have used fuel of two size fractions 600/800 μ m and 40/60 μ m. Smear density was in the range 76 to 77 percent theoretical. Plutonium content of the heavy metal is approximately 15 percent.

- 4836 IRRADIATION EXPERIMENTS OF URANIUM CARBIDE, CARBONITRIDE, AND NITRIDE FUELS IN JAERI. Watanabe, H.; Kurihara, M.; Shimokawa, J.; Ichikawa, M.; Uchida, M. (Japan Atomic Energy Research Inst., Tokai). pp 361-374 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

In JAERI, the irradiation test and analysis have been made to provide information on in-pile behavior of uranium carbide, carbonitride and nitride fuels as a preceding stage examination for the mixed uranium-plutonium fuels. Five capsules containing UC, UC/sub x/N/sub 1-x/(x = 0.75, 0.5 and 0.25) and UN fuel pins were irradiated in JMTR (Japan Material Testing Reactor) to investigate their dimensional changes and swelling rates, and to examine the compatibility of the fuel pellets with the cladding materials under sodium or helium environment.

- 4837 U.S. ADVANCED FUELS SAFETY PROGRAM. Rose, D. (Argonne National Lab., IL); Opendaker, A. pp 377-388 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

An advanced fuels safety program plan has been developed to provide a coordinated approach to resolving the safety issues arising from the use of the candidate advanced fuels (carbide/nitride) in LMFBRs. The advanced fuels safety program plan takes cognizance of the safety issues that have been identified for oxide-fueled LMFBRs, and of the safety technology required to resolve these issues. Specific additional technology needs are identified for advanced fuels. Thus, the program of work proposed builds on the similarities with work already in progress in the oxide program and focuses on the differences or issues unique to advanced fuels.

- 4838 EXPERIMENTAL STUDY OF ADVANCED FUEL-PIN FAILURE PHENOMENA. Wrona, B.J.; Galvin, T.M. (Argonne National Lab., IL). pp 421-432 of

Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

A direct-electrical-heating apparatus was used to perform fundamental fuel-motion experiments on unirradiated, clad and unclad UC pellets to investigate the response of the fuel to simulated in-reactor transient heating rates and power levels. The results show that the response of UC to applied power transients is significantly different than UO_2 behavior in similar situations. Because of the large temperature interval between the boiling and melting points of UC and the high thermal conductivity of UC that results in relatively flat radial temperature gradients, two regimes of gross fuel motion can occur, depending upon the fuel centerline temperature.

4839 GFEK EXPERIMENTAL APPROACH TO CARBIDE FUEL ELEMENT LICENSING. Plitz, H. (Gesellschaft fuer Kernforschung, Karlsruhe, Ger.). pp 433-454 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Information is presented concerning the general approach to licensing the reference carbide fuel elements; design basis of the reference carbide fuel element concept; theoretical and feasible pin design aspects; irradiation statistics, experimental failure limits; incidental behavior and safety aspects; and experimental approach to licensing the reference carbide fuel element.

4840 CASE FOR FUEL REDESIGN AFTER IN PILE SAFETY EXPERIMENTS: THE "COMMANDO" TESTS. Chagrot, M. pp 455-467 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

The CEA reference design for LMFBR advanced fuel elements consisted of mixed carbide sintered pellets stacked in vented, stainless steel clad pins with a large liquid sodium bonded gap. For clearing up the uncertainties of pin behavior, in the case of bond blow-out, two tests were performed in the SCARABEE loop. This experimental program, named COMMANDO, although limited, yielded satisfactory results. The tests showed that no damageable cladding melting occurred and that a new workable steady-state was almost immediately recovered. From these findings and, although other concepts are now considered as well, a safer pin design was devised.

4841 CORRELATION BETWEEN PROPERTIES AND IRRADIATION BEHAVIOR FOR CARBIDE, CARBONITRIDE AND NITRIDE FUELS. Blank, H. (Joint Research Centre, Karlsruhe, Ger.). pp 482-501 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

A description is presented of the general fuel and pin behaviour on the basis of the presently available information of the project and is organized in the following way: the results of an analysis of the complex system

reactor-cladding-fuel in order to fix the parameters and their range of values in the problem of swelling; a description of the general pin behaviour follows as obtained from intermediate and final measurements of the permanent diameter increases ($\Delta D/D$) of the fuel pins; a detailed description of fuel restructuring and its correlation with the basic fuel properties; and the results are summarized and conclusions drawn about the potential of the various types of fuels and pins with regard to fuel composition.

4842 IRRADIATION PERFORMANCE ANALYSIS OF UK FAST REACTOR FUEL. Bellamy, R.G.; Hayns, M.R.; Matthews, J.R.; Wilmore, D. pp 502-515 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Two theoretical fuel element models are used in the UK fast reactor fuel program. SLEUTH was originally developed to study gas cooled thermal reactor problems, while FRUMP was specifically developed for operational and safety studies of fast reactor fuel element behaviour.

4843 PROGRESS IN MODELING CARBIDE AND NITRIDE FUEL PERFORMANCE IN ADVANCED LMFBRs. Billone, M.C.; Jankus, V.Z.; Kramer, J.M.; Yang, C.I. (Argonne National Lab., IL). pp 516-536 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

The UNCLE-S and UNCLE-T codes have been developed to predict the steady-state and transient performance of (U,Pu)C and (U,Pu)N fuels in advanced LMFBRs. The fuel models and properties that have been incorporated into those codes and the comparison between code predictions and postirradiation data are reviewed. The need for further model development in the areas of fuel cracking and healing, fuel melting, fission-gas release and swelling, and high strain-rate fuel deformation are established. An improved model for fuel cracking is proposed.

4844 ANALYTICAL COMPARISON OF THE IRRADIATION BEHAVIOR OF FAST REACTOR CARBIDE AND OXIDE FUEL PINS. Nayak, U.P.; Boltax, A.; Skalka, R.J.; Biancheria, A. (Westinghouse Electric Corp., Pittsburgh). pp 537-561 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Advanced breeder reactor system design studies indicate that mixed U,Pu carbide fuel provides potential advantages over mixed U,Pu oxide fuel related to nuclear, fuel cycle, and safety considerations. However, the state-of-the-art technology on carbide fuel lags the large world-wide effort on oxide fuel. An analytical comparison of the two fuel types under steady-state and transient conditions using the current versions of the LIFE fuel performance codes is presented.

4845 LIFE CODE SYSTEM FOR ANALYSIS OF OXIDE FUEL PIN THERMAL/STRUCTURAL BEHAVIOR. Stephen, J.D.; Biancheria, A.; Billone, M.C.

pp 562-581 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

The LIFE development program is providing national methods for thermal and mechanical analysis and design of FBR fuel pins. The program has produced the LIFE3 code and is developing LIFE4 and LIFE4-transient for (U,Pu)O₂, has encouraged the spinoff development of the UNCLE code for (U,Pu) Carbide and Nitride, and is in the early stages of development of codes for UO₂ (Th,U)O₂, ThO₂ and other blanket and alternate fuel concepts.

- 4846 DEVELOPMENT OF LMFBR CORE DESIGNS USING REFERENCE AND ADVANCED OXIDE FUEL. Chandler, J.C.; Marr, D.R.; McCurry, D.C.; Cantley, D.A.; Omberg, R.P. (Hanford Engineering Development Lab., Richland, WA). pp 583-593 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

The design of an LMFBR core is a complex process which is strongly dependent upon design philosophy, design criteria, selection of material properties, and the calculational procedure involved. The development of a core design with optimum characteristics, whether it be doubling time, power costs, or safety characteristics, requires a close coupling of the disciplines of reactor physics, thermohydraulics, mechanical design, and fuel pin performance. This, by necessity, forces the development of a closely-coupled multidiscipline approach. This is true regardless of whether the fuel is oxide, carbide, nitride, or metal, and regardless of whether the fuel cycle is Pu/U or U-233/Th.

- 4847 ASSUMPTIONS, CONSTRAINTS AND METHODOLOGY OF ANL ADVANCED FUELS SYSTEM STUDIES. Barthold, W.P.; Lam, S.K.; Orehwa, Y. (Argonne National Lab., IL). pp 594-613 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

A preliminary assessment has been made at ANL of the breeding performance of carbide and nitride fuels in 2000 MWe LMFBRs, and its sensitivity to changes in design and performance parameters. In addition, various design trade-off studies, as well as special problems connected with the design of large LMFBRs, were investigated. The preliminary nature of this assessment is a reflection of the lack of data related to the irradiation behavior of these fuels. Both types of fuel were either sodium- or helium-bonded to the cladding. The assumptions and constraints used throughout the study and the methodology employed for the performance and design analysis are summarized. Typical results are given as examples.

- 4848 OPTIMIZATION SYSTEM DESIGN STUDIES OF ADVANCED CARBIDE FUEL. Caspersson, S.A.; Noyes, R.C.; Kulwich, M.R.; diLauro, G.F.; Klinetob, R.H. (Combustion Engineering, Inc., Windsor, CT). pp 614-633 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

System design studies of advanced helium and sodium-bonded uranium-plutonium carbide fuel in commercial LMFBR's have been completed to develop recommendations for optimum values of advanced fuel pin design parameters. Fuel cycle costs, doubling times, and uranium ore requirements were applied as principal optimization criteria. When fuel pin diameter is allowed to vary, higher linear powers result in lower achievable fuel cycle costs and give low doubling times and low uranium ore requirements.

- 4849 INVESTIGATION OF SODIUM-BONDED CARBIDE FUEL ASSEMBLY DESIGN. Madell, J.T.; de Paz, J.F.; Yang, C.I. (Argonne National Lab., IL). pp 634-647 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

The engineering studies consisted of steady-state thermal-hydraulic analysis of the fuel assembly, transient thermal analysis of advanced fuel cores, performance studies of pin spacer types, structural analysis of the duct, flow induced vibration analysis, core resistant analysis and cladding structural lifetime calculations. The results of this study will be incorporated into a new assembly design.

- 4850 DESIGN AND PERFORMANCE OF SHROUD TUBES. Kerrisk, J.F.; Barner, J.O.; Petty, R.L. (Los Alamos Scientific Lab., NM). pp 648-659 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Shroud tubes were conceived as a means of preventing the rearrangement and wedging of broken fuel pieces in sodium-bonded advanced-fuel elements. The specific design of a thin-walled tube with slots that fits closely around the fuel was arrived at by consideration of a number of competing mechanical and heat-transfer requirements. Postirradiation examination results obtained to date indicate that shroud tubes are meeting the design requirement of retaining fuel fragments. No cladding failures have been observed in any sodium-bonded element using a shroud tube. The excellent irradiation performance of sodium-bonded elements with shroud tubes as compared to similar elements without shroud tubes demonstrates the utility of the shroud-tube concept.

- 4851 STATUS OF INTERNATIONAL FUEL CLADDING DEVELOPMENT. Laidler, J.J. (Hanford Engineering Development Lab., Richland, WA). pp 681-694 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Several substantial national and international programs on cladding and duct materials development are currently in progress, and great strides are being made in characterizing the behavior of current reference materials and in developing materials with improved performance capabilities. A brief overview of these programs, the state of materials development, and the nature of some

current important issues pertaining to materials behavior in the reactor environment are presented.

- 4852 THERMAL-HYDRAULIC EVALUATION OF ADVANCED WIRE-WRAP ASSEMBLIES. Wei, J.P. (GE, Sunnyvale, Calif.). AICHE Symp. Ser.; 73: No. 164, 105-113(1977).

Fuel assemblies for liquid-metal-cooled fast breeder reactors consist of a large number of rods uniformly spaced by wire wrap in a hexagonal array. Thermal-hydraulic analyses of a 217-rod wire-wrapped assembly with different wire-wrap geometries was accomplished. Advanced designs considered are staggered starts, distributed wireless fuel rods and locked wrap. 10 refs.

- 4853 EMPIRICAL METHOD TO CALCULATE CLINCH RIVER BREEDER REACTOR PLANT (CRBRP) INLET PLENUM TRANSIENT TEMPERATURES. Howarth, W.L. (Westinghouse Electr Corp, Madison, Pa.). AICHE Symp. Ser.; 73: No. 164, 114-119(1977).

Sodium flow enters the CRBRP inlet plenum via three inlet loops. An empirical equation was formulated to calculate transient temperatures in the inlet plenum from known inlet loop flows and temperatures. The constants in the empirical equations were measured in a (one quarter) scale Inlet Plenum Model using water as the test fluid and an electrolyte to simulate the sodium temperature distribution. Step electrolyte transients at 100% model flow were used to calculate the equation constants. The empirical equation was checked by comparing its results with test results from additional tests that covered the inlet plenum operational range. 4 refs.

- 4854 INFLUENCE OF SCALE SIZE AND FLUID THERMAL PROPERTIES IN SIMULATING LMFBR OUTLET PLENUM BEHAVIOR. Lorenz, J.J.; Carlson, R.D.; Howard, P.A. (Argonne Natl Lab, Ill.). AICHE Symp. Ser.; 73: No. 164, 120-126(1977).

The influence of scale size and fluid thermal properties in simulating LMFBR outlet plenum behavior was studied. Tests were performed with water and sodium using 1/15-scale models of LMFBR outlet plenums. It was found that differences in the behavior of sodium and water were relatively modest despite a large variance in the respective Peclet numbers. The effect of scale size was found to be unimportant as evidenced by the reasonably good overall agreement of the present 1/15-scale results and 0.55-scale data of J.A. Carr and L.J. Flanigan. It was concluded that for the purpose of obtaining engineering design data relating to LMFBR outlet plenums, small scale water tests will provide a fairly good direct simulation of full scale sodium behavior. 3 refs.

- 4855 INTERMEDIATE HEAT EXCHANGER DEVELOPMENT FOR THE CLINCH RIVER BREEDER REACTOR PLANT. Sawyer, R.H.; Cho, S.M.; Pai, D.H. (Foster Wheeler Energy Corp, Livingston, NJ). Am. Soc. Mech. Eng., [Pap.]; No. 77-WA/NE-10, vp(1977).

The design and development of the Intermediate Heat Exchanger (IHx) for the Clinch River Breeder Reactor Plant (CRBRP) has progressed from its performance sizing stage through flow testing of critical areas to its final performance analysis. The flow testing provided the impetus for the design evolution of the flow distribution and flow control devices throughout the IHx. The design evolution of the IHx was also shaped by factors, such as manufacturing economics, the

manufacturing experience of the Fast Flux Test Facility (FFTF) IHx, the structural considerations to meet the current code requirements, and flow-induced dynamic considerations within the bundle. These are but some of the areas discussed in this paper. 3 refs.

- 4856 DEVELOPMENT OF LMFBR STEAM GENERATOR LEAK PROTECTION SYSTEMS. Magee, P.M.; Gerrels, E.E.; Greene, D.A.; McKee, J. (GE, Sunnyvale, Calif.). Am. Soc. Mech. Eng., [Pap.]; No. 77-WA/NE-12, vp(1977).

Leak protection for Liquid Metal Fast Breeder Reactor steam generators is an important economic factor in the shutdown, repair and restart of a plant. Development of leak protection systems in the U.S. has concentrated on four areas: chemical (H_2 and O_2) leak detection meters; acoustic leak detection/location techniques; investigation of leak behavior (enlargement, damage effects, plugging and unplugging); and data management for plant operations. This paper discusses the status, design aspects, and applications of leak protection technology for LMFBR plants. 11 refs.

- 4857 TESTING OF FFTF FUEL HANDLING EQUIPMENT. Coleman, D.W.; Grazzini, E.D.; Hill, L.F. (US ERDA, Richland, Wash.). Am. Soc. Mech. Eng., [Pap.]; No. 77-WA/NE-9, vp(1977).

The Fast Flux Test Facility has several manual/computer controlled fuel handling machines which are exposed to severe environments during plant operation but still must operate reliably when called upon for reactor refueling. The test programs for two such machines--the Closed Loop Ex-Vessel Machine and the In-Vessel Handling Machine--are described. The discussion centers on those areas where design corrections or equipment repairs substantiated the benefits of a test program prior to plant operation. 4 refs.

- 4858 DEVELOPMENT OF SMALL SODIUM VALVES FOR THE FAST FLUX TEST FACILITY. Verderosa, M.A.; Bazinet, G.D. (US ERDA, Richland, Wash.). Am. Soc. Mech. Eng., [Pap.]; No. 77-WA/NE-13, vp(1977).

The FFTF utilizes approximately 250 small (10.2 cm or smaller) sodium valves in its auxiliary liquid metal systems. This paper traces the design evolution of the two valve concepts chosen along with the associated fabrication, inspection and testing histories focusing on the problems identified and the solutions employed. The post-test examinations of the valves are reviewed and the actual valve capabilities versus the original design goals are discussed. Recommendations for future valve development, including potential cost reduction schemes, are identified. 3 refs.

- 4859 EXTINCTION OF METAL FIRES. Mellottee, H. (Centre National de la Recherche Scientifique, 45 - Orleans-la-Source (France). Centre de Recherches sur la Chimie de Combustion et des Hautes Temperatures). Rev. Gen. Therm.; 16: No. 186, 527-538(1977). (In French).

The main points of a large bibliography on liquid and solid metal fires are set out. The various methods used to fight these fires are presented; covering by powders is specially emphasized. Since this method has promising results, the various possible techniques, extinction by cooling the metal, by blanketing, by formation of a continuous insulating layer (by fusion or pyrolysis of a powder) or by a

surface reaction between powder and metal are studied. The conditions of conservation and use of powders are outlined, then the various powders are described: inert powders, powders undergoing a physical transformation (fusion or vitrification of an organic compound, fusion of eutectic inorganic mixtures), multiple effect powders. Precise examples are quoted.

- 4860 ANALYSIS OF CORE DISRUPTIVE ACCIDENTS IN CARBIDE-FUELED LMFBRs. Mync, L.; Bowers, C.H.; Ferguson, D.R. (Argonne National Lab., IL). pp 389-406 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Parametric studies of two hypothetical core disruptive accidents in a large carbide-fueled LMFBR are discussed. The SAS3A code is used to produce speculative scenarios for the unprotected loss-of-flow and transient overpower accident initiators. Aside from the treatment of transient phenomena unique to carbide fuel, the predictions are based mainly on a direct extrapolation from CDA phenomenology developed for oxide fuels. In addition, a comparison is made to the predicted behavior of a similarly-sized oxide-fueled LMFBR to these two accident initiators.

- 4861 SAFETY ASPECTS OF CARBIDE IRRADIATIONS IN FTR DRIVER POSITIONS. Nickerson, G.M.; Bergeron, R.J.; Siegmund, E.R. (Combustion Engineering, Inc., Windsor, CT). pp 407-420 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

An assessment has been made of safety implications for irradiation testing of advanced LMFBR carbide fuel in the Fast Test Reactor (FTR). Carbide fuel has been compared with FTR reference oxide fuel; findings indicate that carbide fuel can be irradiated in FTR safely. Investigations were made of two hypothetical core disruptive accident initiating transients: the transient overpower and the transient under cooling sequences. Improved modeling of sodium bond and shroud tube behavior has been developed. Events are changed little by including parametric variations in sodium bond and shroud tube behavior.

- 4862 ADVANCED FUELS SAFETY COMPARISONS. Grohmes, M.A. (Argonne National Lab., IL). pp 469-480 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

The safety considerations of advanced fuels are described relative to the present understanding of the safety of oxide fueled Liquid Metal Fast Breeder Reactors (LMFBR). Safety considerations important for the successful implementation of advanced fueled reactors must early on focus on the accident energetics issues of fuel coolant interactions and recriticality associated with core disruptive accidents. It is in these areas where the thermal physical property differences of the advanced fuel have the greatest significance.

- 4863 MEASUREMENTS OF SUBASSEMBLY AND CORE TEMPERATURE DISTRIBUTIONS IN AN LMFBR. Singer, R.M.; Gillette, J.L. (Argonne Natl Lab, Ill). AICHE Symp. Ser.; 73: No. 164, 97-104(1977).

In the analysis of temperature distributions in the core of a liquid metal cooled fast breeder reactor (LMFBR) under both normal and abnormal operating conditions, it is generally assumed that there is little, if any, thermal interaction between adjacent subassemblies. Since intersubassembly heat transfer tends to reduce the radial temperature gradient in a reactor, thereby ameliorating the effects of local overheating, this assumption is conservative. In order to assess the importance of this effect as well as that of flow redistribution in a reactor core, a series of in-core and subassembly outlet measurements were made in EBR-II over a wide range of operating power and flow conditions, encompassing both forced and natural convection. The results of this study indicate that radial heat transfer and flow redistribution are important mechanisms which should be taken into account in the thermal-hydraulic analyses of LMFBR cores, especially under low flow rate conditions. 14 refs.

- 4864 SOME OBSERVATIONS ON OXYGEN AND CARBON POTENTIALS IN U-Pu-O AND U-Pu-C SYSTEMS. Tetenbaum, M. (Argonne National Lab., IL). pp 179-188 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Measurements to characterize the effect of plutonium valency on oxygen potential have been made at 2050 K for hypostoichiometric U-Pu-O fuel compositions having plutonium-to-metal ratios ranging from 0.15 to 0.30. The data indicate that oxygen potential is dependent on the plutonium-to-metal ratio and therefore, do not support the hypothesis that $\Delta G(O_2)$ values for the U-Pu-O system are a function of the metal whose oxidation state is changing. Equilibrium of the U-Pu-C solid solution system, where $Pu/(U + Pu) = 0.2$ and $C/(U + Pu) = 0.98$ with a carrier gas having a carbon activity of approximately 0.62 at 2050 K resulted in the formation of a residue of composition $C/(U + Pu) = 1.05$.

- 4865 THERMOPHYSICAL PROPERTIES OF LIQUID URANIUM CARBIDE. Finn, P.A.; Sheth, A.; Winslow, G.; Leibowitz, L. (Argonne National Lab., IL). pp 189-204 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Uranium-plutonium carbide, which has better neutronic properties and higher thermal conductivity than uranium-plutonium oxide, has been considered as a fuel for nuclear reactors. In order to perform essential reactor safety analyses, many thermodynamic and transport properties at temperatures exceeding 6000 K are needed. The required values include vapor pressure, density, volume expansion, heat capacity, enthalpy, compressibility, and viscosity. It was necessary to develop a self-consistent method that could be used to extrapolate known property values to the liquid region. This report summarizes work on uranium carbide.

- 4866 FUEL PROPERTIES OF ADVANCED FUELS (U,Pu)CN IMPORTANT FOR OUT-OF-PILE AND IN-PILE KINETICS. Matzke, H.; Ronchi, C. (Joint Research Centre, Karlsruhe, Ger.). pp 218-244 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

The investigated fuels covered the whole range of (U,Pu)C over (U,Pu)C/sub x/N/sub 1-x/ to (U,Pu)N, and some additional work was devoted to oxycarbides (U,Pu)CO. The properties studied are described and include self-diffusion, sintering and densification, mechanical properties like plasticity and crack formation, thermal conductivity, pore mobility and fuel restructuring in an applied thermal gradient simulating expected in-pile thermal gradients, and solubility of fission products.

- 4867 CREEP BEHAVIOR OF MIXED-CARBIDE ADVANCED NUCLEAR FUEL. Singh, R.N. (Argonne National Lab., IL). pp 257-266 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

Primary and steady-state creep behavior of (U_{0.8}Pu_{0.2})C_{1.0} advanced nuclear fuel was investigated as functions of stress (6.89 to 68.9 MNm⁻²) and temperature (1573 to 1873 K). Mixed carbide fuel exhibits higher primary and steady-state creep rates when compared with stoichiometric mixed-oxide fuel of the same density, grain size, and Pu concentration for identical testing conditions.

- 4868 CHEMICAL STATE OF IRRADIATED CARBIDE FUEL PINS. Kleykamp, H. (Kernforschungszentrum, Karlsruhe, Ger.). pp 166-178 of Advanced LMFBR fuels. Leary, J.; Kittle, H. (eds.). La Grange Park, IL; American Nuclear Society (1977).

From International meeting on advanced LMFBR fuels; Tucson, AZ, USA (10 Oct 1977).

See CONF-771008--.

The postirradiation studies of uranium-plutonium carbide fuel pins for the development of a fuel element for operation in the German SNR-300 reactor have enabled characterization of the chemical state of carbide fuels. The latter is influenced by the compositional changes during irradiation resulting from the fission process and by redistribution of the fuel components and the fission products. These effects have been investigated by metallographic techniques, electronprobe microanalysis, and scanning electron microscopy on fuel pins that had been irradiated in the thermal reactors FR2 and BR2 and in the Dounreay Fast Reactor DFR. The current status of knowledge based on these irradiation experiments is reviewed.

- 4869 EDDY-CURRENT EXAMINATION OF IRRADIATED FUEL ELEMENTS AT THE HOT FUEL EXAMINATION FACILITY. Bhipps, R.D.; Kleiss R.E.; Coughlan, J.B. (Argonne National Lab., Idaho Falls, ID). pp 245-250 of Proceedings of the 25th conference on remote systems technology. Wojciechowski, D.R. (ed.). La Grange Park, IL; American Nuclear Society, Inc. (1977).

From 25. remote systems technology symposium; San Francisco, CA, USA (27 Nov 1977).

See CONF-771117--.

The use of eddy-current examinations on irradiated fuel cladding is a promising but

relatively undocumented testing method. At the Hot Fuel Examination Facility, an eddy-current cladding-integrity tester is being used to better understand the effects of radiation on fuel pins. The goal is a high throughput system that will test for microscopic flaws, cracks, or unusual defects that might be impossible to detect by any other available means.

- 4870 FUEL HANDLING AND STORAGE SYSTEM FOR LARGE COMMERCIAL LMFBRs. Davis, C.R.; Sciaccia, F.W. (General Electric Co., Sunnyvale, CA). pp 386-396 of Proceedings of the 25th conference on remote systems technology. Wojciechowski, D.R. (ed.). La Grange Park, IL; American Nuclear Society, Inc. (1977).

From 25. remote systems technology symposium; San Francisco, CA, USA (27 Nov 1977).

See CONF-771117--.

A fuel handling and storage system suitable for large commercial liquid-metal fast breeder reactor plants has been conceptually developed and designed. The system provides for all in-plant handling and storage of core components. The particular arrangements and features were chosen to provide rapid refueling capability, a highly reliable and safe system, a cost effective system, and a system that requires a minimum of research and development effort.

- 4871 ANALYSIS OF ERDA'S NATIONAL ENERGY PLAN. Coccio, C.L. (GE, Schenectady, NY). Am. Soc. Mech. Eng., [Pap.]; No. 77-WA/TS-2, vp(1977).

This national energy plan begins with five overall policy goals as follows: (a) maintain the security and policy independence of the nation, (b) maintain a strong and healthy economy, (c) provide for future needs so that future life styles are not limited by energy unavailability, (d) contribute to world stability, and (e) protect and improve the nation's environmental quality. Within this general framework, ERDA develops eight technology goals such as "increase the use of essentially inexhaustible domestic energy resources." Programs are then identified under each goal such as solar electric, breeder reactors, and fusion for the example above. A review of the plan shows that some of the more significant items called for in the legislation relating to anticipated program results, costs, and impacts were not described. When Congress became aware of this lack of quantification, letters were sent to ERDA requesting specific responses along the above lines. This paper presents ERDA's subsequently furnished program costs and benefits in the eight technology goal categories of the national energy plan.

- 4872 NUCLEAR PROLIFERATION AND NUCLEAR POWER. Woodruff, G.L. (Univ of Wash, Seattle). Am. Soc. Mech. Eng., [Pap.]; No. 77-WA/TS-1, vp(1977).

In April 1977, President Carter announced that the United States would defer indefinitely the commercial reprocessing of nuclear fuel and the commercialization of the liquid metal fast breeder reactor. These decisions were made out of concern for nuclear proliferation. It is very unlikely that even a simple majority of nations will forego the long-term benefits of reprocessing and breeders. At the time of this writing, not even the U.S. Congress seems inclined to accept these aspects of the President's program. International controls of enrichment and reprocessing are more-promising alternatives as proliferation safeguards. A number of safeguards approaches may be highly effective. For example, separated plutonium can

be "spiked" with radioactivity and thereby rendered relatively safe from theft by terrorists. Satellite monitoring of reactor operations can provide instant detection of attempts to divert fuel.

- 4873 FAST FLUX TEST FACILITY CORE COMPONENT MEASURING SYSTEM. Mullen, R.E. (Aerojet Manufacturing Co., Fullerton, CA); Harbison, S.G.; Martin, R.L. pp 67-77 of Proceedings of the 25th conference on remote systems technology. Wojciechowski, D.R. (ed.). La Grange Park, IL; American Nuclear Society, Inc. (1977).

From 25. remote systems technology symposium; San Francisco, CA, USA (27 Nov 1977).

See CONF-771117--.

Aerojet Manufacturing Company has designed and procured a core component measuring system for installation in the Fast Flux Test Facility (FFTF) interim examination/maintenance cell. This remotely operable and maintainable system will define the geometric changes that are expected to occur in FFTF core components by measuring cross sections, length, bow, and twist. The system will be used to measure the external temperature at a specific axial position along the hex duct, differential pressure across a core component, and axial flow rates of cooling gas through a component. Full automation of the measuring system makes it possible to acquire, process, and record vast amounts of data with minimal operator attention. Data printouts, historical comparison, distribution plots, and listings for irradiation experiments provide immediate and useful data to the sponsors of FFTF experiments.

- 4874 FAST FLUX TEST FACILITY ELEMENT MEASURING SYSTEM. Mullen, R.E. (Aerojet Manufacturing Co., Fullerton, CA); Harbison, S.G.; Martin, R.L. pp 78-89 of Proceedings of the 25th conference on remote systems technology. Wojciechowski, D.R. (ed.). La Grange Park, IL; American Nuclear Society, Inc. (1977).

From 25. remote systems technology symposium; San Francisco, CA, USA (27 Nov 1977).

See CONF-771117--.

Aerojet Manufacturing Company has been responsible for the overall design and procurement of an element measuring system to be installed in the Fast Flux Test Facility (FFTF) interim examination/maintenance cell. The system is remotely operable and maintainable and will be used to detect both geometric and weight changes occurring in FFTF elements. In addition, the system will be used to determine fuel pellet movement, gamma flux profiles, and isotope migrations, and to detect microscopic cladding failures. Fully automated computer control makes it possible to process and acquire data from up to 24 elements with minimal operator attention.

- 4875 DESIGN FEATURES AND OBJECTIVES OF THE FUELS AND MATERIALS EXAMINATION FACILITY. Doman, D.R.; Trego, A.L.; Burgess, C.A. (Westinghouse Hanford Co., Richland, WA). pp 90-102 of Proceedings of the 25th conference on remote systems technology. Wojciechowski, D.R. (ed.). La Grange Park, IL; American Nuclear Society, Inc. (1977).

From 25. remote systems technology symposium; San Francisco, CA, USA (27 Nov 1977).

See CONF-771117--.

The conceptual design has been nearly completed on the Fuels and Materials Examination Facility (FMEF), a hot-cell facility for destructive (DE) and

nondestructive examination (NDE) of irradiated fuels and materials tested in the Fast Flux Test Facility (FFTF) and other breeder reactors. The FMEF will be constructed near the FFTF and is scheduled to be operational by early 1983. A large cell with 22 work stations permits NDE of fuel assemblies and pins, while 12 smaller cells provide sectioning of failed assemblies or pins and DE. These cells contain a recirculating nitrogen atmosphere. To handle the large assemblies and increased number of fuels and materials specimens irradiated in the FFTF, the facility is highly mechanized to provide high throughput.

- 4876 ADVANTAGES OF EQUIPMENT TESTING IN THE IEM CELL MOCKUP. LeGore, T.; Dietrick, D.L.; Engstrom, J.W.; Richman, R.B. (Westinghouse Hanford Co., Richland, WA). pp 130-133 of Proceedings of the 25th conference on remote systems technology. Wojciechowski, D.R. (ed.). La Grange Park, IL; American Nuclear Society, Inc. (1977).

From 25. remote systems technology symposium; San Francisco, CA, USA (27 Nov 1977).

See CONF-771117--.

The Fast Flux Test Facility (FFTF) interim examination and maintenance cell will contain a large amount of equipment for handling FFTF core components. To adequately assure that the equipment is operational prior to installation at FFTF, a full-scale mockup was built and an extensive test program was carried out. The tests covered installation, remote maintenance, and systems integration of more than eighty different equipment items to be used in the cell.

- 4877 SPENT-FUEL SHIPPING SYSTEM FOR LARGE COMMERCIAL LMFBRs. Davis, C.R. (General Electric Co., Sunnyvale, CA). pp 397-402 of Proceedings of the 25th conference on remote systems technology. Wojciechowski, D.R. (ed.). La Grange Park, IL; American Nuclear Society, Inc. (1977).

From 25. remote systems technology symposium; San Francisco, CA, USA (27 Nov 1977).

See CONF-771117--.

A spent-fuel shipping system for large commercial liquid-metal fast breeder reactors (LMFBRs) has been conceptually developed and the major components designed. The system achieves double containment by seal welding individual spent-fuel assemblies into a shipping canister and sealing several canisters into a shipping cask with a dry cavity. The cask is loaded without removing it from its transporter and without contaminating its exterior surfaces. Early adoption of this (or an equivalent) spent-fuel shipping system is needed to provide guidance to the designers of LMFBR equipment and facilities, as well as of the LMFBR fuel reprocessing plant.

- 4878 NUCLEAR CRISIS. A QUESTION OF BREEDING. Gosling, D.; Montefiore, H. (eds.). Dorchester, Dorset; Prism Press (1977). 174p.

This book contains an edited account of the submissions and the cross-examination of Expert Witnesses at the Public Hearings on the projected Commercial Fast Reactor CFR-1 which were held at the London International Press Centre on 13 and 14 December 1976. The account is in chapters dealing with different aspects of the matter: (1) statement by Mr. Benn (Secretary of State for Energy) and replies; (2) future needs and goals for society; (3) risks (technological and social); (4) feasibility, economic and technological; (5) CFR-1 and alternative sources of energy; (6) epilogue.

- 4879 **QUANTITATIVE ASSESSMENT OF NUCLEAR INCINERATION OF ACTINIDE WASTE.** Harte, G.A.; Clarke, R.H. (Central Electricity Generating Board, Berkeley (UK). Berkeley Nuclear Labs.). pp 289-292 of Proceedings of the 4. International congress of the International Radiation Protection Association. Paris, 24-30 April 1977. Vol. 2. Fontenay-aux-Roses, France; Association Internationale de Protection contre les Rayonnements (1977).
From 4. congress of the International Radiation Protection Agency; Paris, France (24 Apr 1977).

See CONF-770409--. Available from M. Gilbert Bresson, General Secretary IRPA, B.P. No. 33, 92260 Fontenay-aux-Roses, France.

The actinide arisings from reactors currently of interest in Europe, both thermal and fast, are compared over cooling times from 10^0 to 10^6 years by use of an index of biological hazard. The effects of reirradiation of this waste are considered in both light water (PWR) and sodium cooled (LMFBR) commercial reactor designs, again in terms of the change in biological hazard index over long cooling times. The criterion for judging the effectiveness of nuclear incineration is proposed as the hazard index of the amount of ^{238}U plus its equilibrium daughter activities that is burned up in the primary reactor fuel cycle. This criterion enables an upper limit to the cooling (decay) time after irradiation which must be considered important for waste disposal. From these studies it is possible to conclude that the current national design of a plutonium fuelled LMFBR provides potential physics conditions for nuclear incineration to be feasible, whereas recycling the waste through a PWR fuel cycle is unlikely to provide any significant benefit. It is further shown that the actinide waste from early reactors is not worth incinerating unless essentially 100% uranium and plutonium removal is achieved before irradiation. If plutonium is used in either thermal or fast reactors, it is shown that U + Pu separation of greater than 99.9% is required before incineration will reduce the ultimate waste hazard at long cooling times and then only after an irradiation time of at least 30 years at peak flux in a commercial LMFBR.

- 4880 (KFK-tr--536) **EXPERIMENTAL RESULTS WITH CARBIDE FUEL RODS IN THE BOR-60 REACTOR.** Tsykanov, V.A.; Gryazev, V.M.; Davydov, E.F.; Kuz'min, V.I.; Malershin, A.A.; Syuzev, V.N.; Golovnin, I.S.; Men'shikova, T.S.; Bibilashvili, Yu.K.; Kotel'nikov, R.B. Translated from At. Energ. (USSR); 42: No. 5, 378-382(1977). 17p. (In German). Dep. NTIS (US Sales Only), PC A02/MF A01.

Translated from Russian: Rezul'taty ispytaniya v reaktore BOR-60 tvelov s karbidnym toplivom.

In recent research work, much attention has been paid to the study of carbide fuels for fast power reactors, which have a better thermal conductivity and a larger number of heavy atoms than conventional oxide fuels. This is due to the fact that they may help to shorten the doubling time, to increase the breeding rate and thus to improve the efficiency of fast reactors. The present paper gives some results of a study on the behaviour of irradiated carbide fuels and fuel rods which had been in constant operation in the BOR-60 reactor up to a burnup of 10% of the heavy atoms.

- 4881 **LEAK DETECTION OF STEAM OR WATER INTO SODIUM IN STEAM GENERATORS OF LIQUID-METAL FAST**

BREEDER REACTORS. Hans, R.; Dumm, K. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). At. Energy Rev.; 15: No. 4, 611-699(1977).

The leakage of water or steam into sodium in LMFBR steam generators, including a study of how leaks are detected and located as well as the potential damage that could be caused by such leaks, is surveyed. The most interesting steam generator designs evolving in those countries that develop and construct LMFBRs are presented. The relevant protection measures are described. Fault conditions are defined and descriptions given of possible sequences of events leading to abnormal conditions in a steam generator. Taking into account theory, the potential of the hydrogen and oxygen detection systems is discussed. Different hydrogen and oxygen detection systems are fully described. In so far as interesting technical solutions are concerned, previously developed devices have also been taken into account. The way oxygen detection supplements hydrogen detection is described by listing the available oxygen measuring devices and the relevant theory. Only a few sonic and accelerometer measurements have been made on complete steam generator units so there is little system data available. Descriptions, however, have been included to give the state of the art achieved for the sensors and the achieved sensitivities or band widths. The potential of this monitoring method is made evident by adding the technical data of the sensors. Furthermore, the available systems for monitoring medium and large leakages are described. Finally, recommendations are made concerning steam generator development and the application of hydrogen and oxygen detection systems, as well as acoustic measuring methods for small-leakage detection.

- 4882 **FLOW IN THE INLET PLENUM FEATURE MODEL OF THE CLINCH RIVER BREEDER REACTOR.** McConnell, P.M. (Westinghouse Hanford Co, Richland, Wash.). Am. Soc. Mech. Eng., [Pap.]; 77-WA/FE-19: vp(1977).

Results of the Inlet Plenum Feature Model (IPFM) hydrodynamic design verification tests are presented. The IPFM models all hydrodynamically important wetted surfaces in a full 360° (one quarter scale) sector of the inlet plenum and lower internal components of the Clinch River Breeder Reactor (CRBR). These tests were aimed at establishing a near optimum inlet plenum configuration; determining the fluid dynamic performance of the current CRBR auxiliary flow port liner design; evaluating the susceptibility of the inlet plenum design to flow blockage from various types of hypothetical debris; and assessing the ability of the inlet plenum to break up hypothetical bubbles injected into the inlet piping. The test results indicate that the CRBR inlet plenum will produce flow maldistributions of less than ± 3 percent and that mixing of the liquid sodium coolant will not be appreciably affected by substantial changes in the plenum height or the reactor flow rate. 16 refs.

- 4883 **EULERIAN METHOD FOR LARGE-DISPLACEMENT FLUID-STRUCTURE INTERACTION IN REACTOR CONTAINMENTS.** Chang, Y.W.; Wang, C.Y. (Argonne Natl Lab, Ill). pp 1-14 of Canadian Nuclear Association--annual international conference proceedings, 17, Volumes 1 through 6, 1977. Montreal; Canadian Nuclear Association (1977).

From 17. annual international conference of the Canadian Nuclear Association; Montreal,

Ontario, Canada (5 Jun 1977).

An Eulerian method for analyzing large-displacement fluid-structure interaction in reactor containments is presented. The emphasis is on the development of a generalized hydrodynamic scheme to treat the irregular cells created by the movement of the structure with respect to the fixed Eulerian coordinates. A relaxation equation is derived from the boundary condition at the fluid-structure interface for the solution of the pressure at the interface. By combining this with the Poisson equation a sufficient set of equations is obtained for the determination of the advanced-time pressures in the fluid region and at the fluid-structure interfaces. Sample problems are given to illustrate the analysis.

- 4884 STRUCTURAL MECHANICS IN REACTOR TECHNOLOGY. VOL. E. STRUCTURAL DYNAMICS IN FAST REACTOR ACCIDENT ANALYSIS. Jaeger, T.A.; Boley, B.A. (eds.). Brussels; Commission of the European Communities (1977). 404p. (CONF-770807--Pt.E).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This volume covers the sessions on structural dynamics in fast reactor accident analysis. Separate abstracts were prepared for each of the individual papers included.

- 4885 UNITED STATES URANIUM POSITION. Boyd, J. pp 143-151 of Uranium supply and demand. London; Mining Journal Books (1977).

From 2. symposium on uranium supply and demand; London, UK (22 Jun 1977).

See CONF-7706120--.

It is concluded that the time between the decision to mine and equip a given unit of uranium reserves and the time that it will be available to a power plant is 10 to 15 years. The time in which that uranium is in residence in a LWR before the derived plutonium and ^{238}U are recaptured for further use is of the order of 10 or more years. It will take several years to reverse the present prohibition on recycling and reactivate or build the required plants. It appears that, unless drastic action is taken, there will not be sufficient uranium forthcoming to sustain a nuclear industry based upon LWRs. As a consequence, it is imperative that we extend our uranium resources and reserves by developing and commercializing some sort of advanced converter as soon as possible. The present state of the art would indicate that this will have to be the Liquid Metal Breeder Reactor or the advanced system (Candu) adopted by the Canadians.

- 4886 IN-REACTOR DEFORMATION OF SOLUTION ANNEALED TYPE 304L STAINLESS STEEL. Flinn, J.E.; McVay, G.L.; Walters, L.C. (Argonne National Lab., Ill. (USA)). pp 210-223 of Measurement of irradiation-enhanced creep in nuclear materials. Cundy, M.R.; von der Hardt, P.; Loelgen, R.H. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From International colloquium on measurement of irradiation enhanced creep in nuclear materials; Petten, Netherlands (5 May 1976).

See CONF-760530--.

Over the past six years at EBR-II, a great deal of information has been obtained on the in-reactor behavior of solution annealed-Type 304L stainless steel. This information consists of the following: (1) Irradiation induced swelling results in the form of immersion density and transmission electron microscope (TEM) measurements on unstressed material that extends over a temperature range of 395° to

530°C and a neutron fluence range of 1.8 to 9.3×10^{22} n/cm² (E>0.1 MeV). (2) Irradiation induced creep results from helium pressurized capsules irradiated at a temperature of 415°C. The hoop stress range covered in the experiment was 0 to 27.3 ksi, and the peak neutron fluence obtained to date is 7×10^{22} n/cm² (E>0.1 MeV). (3) Residual stress measurements (slit tube technique) with complementary TEM gradient studies on stressed and unstressed capsules. (4) Comparative swelling studies of stressed cladding material and unstressed capsule material from encapsulated EBR-II driver fuel experiments over wide ranges of temperature and neutron fluences. The deformation information derived from the four above studies represent an extensive data base from which to obtain an understanding of the in-reactor deformation of austenitic stainless steel. It is the purpose of this paper to review our information on the in-reactor deformation of solution annealed Type 304L stainless steel.

- 4887 IRRADIATION CREEP DATA IN SUPPORT OF LMFBR CORE DESIGN. Gilbert, E.R.; Straalsund, J.L.; Wire, G.L. (Hanford Engineering Development Lab., Richland, Wash. (USA)). pp 266-278 of Measurement of irradiation-enhanced creep in nuclear materials. Cundy, M.R.; Hardt, P. von der; Loelgen, R.H. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From International colloquium on measurement of irradiation enhanced creep in nuclear materials; Petten, Netherlands (5 May 1976).

See CONF-760530--.

Temperature and neutron flux gradients occurring spatially within the core of an LMFBR generate distortions through the phenomena of swelling and irradiation creep in contemporary core materials. These distortions lead to subassembly interactions both during operation of an LMFBR and during refueling, and thereby limit the life of certain components. Core design and performance analyses therefore require good descriptions of swelling and irradiation creep. Sufficient data to quantify swelling and irradiation creep are finally beginning to materialize. Stainless steel swelling data are available for neutron fluences up to 10^{23} n/cm² (E>0.1 MeV). New testing techniques and analytical methods are being developed to determine the history dependence and stress-state dependence of irradiation creep at both low and high neutron fluence levels to systematically develop the rules required to describe irradiation creep under general multiaxial stress conditions. The initial high fluence results obtained from these new techniques are showing that the limited low neutron fluence, low temperature irradiation creep results available until now did not provide an adequate basis for development of equations which describe irradiation creep for the conditions existing in new experimental or commercial LMFBR systems. These new results are showing significantly more irradiation creep than predicted by extrapolation of the previously available low fluence and low temperature data.

- 4888 INTERNATIONAL CONFERENCE ON IRRADIATION EFFECTS IN STRUCTURAL MATERIALS FOR BREEDER REACTORS. Anderko, K.; Ehrlich, K. (Kernforschungszentrum Karlsruhe (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). KfK Nachr.; 9: No. 2, 60-61(1977). (In German).

From International conference on radiation effects in breeder reactor structural materials; Scottsdale, Ariz., USA (19 - 23 Jun 1977).

A short survey is given of the Scottsdale, USA (1977) conference on irradiation effects in

structural materials for breeder reactors. The following topics are dealt with in more detail: Mechanical properties; in-pile creep, swelling and microstructure, model calculations. It is generally assumed that apart from the basic composition of the metallic materials (Fe, Cr, Ni), small amounts of additives are also of great importance. In most national programmes, this results in a modification of reference materials, investigations of commercial materials with altered composition, and a development of new materials.

- 4889 PREDICTION OF PRIMARY CIRCUIT CONTAMINATION IN POWER REACTORS. Beslu, P.; Pepin, P. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)); Costa, L.; Frejaville, G. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France)). Atomkernenergie; 30: No. 4, 297-302(1977).

From 5. international reactor shielding conference; Knoxville, Tenn. (18 - 22 Apr 1977).

Knowledge of the amount and distribution of activity in the coolant circuit of a power reactor is of great practical importance for determining shielding and for predicting operating personnel radiation exposure. The present status of mathematical models to predict the contamination level in power reactor coolant circuits (High temperature, Fast breeder and Pressurised Water Reactors) are presented in this paper. Some applications are also given.

- 4890 BN-350 REACTOR SHIELDING STUDIES. Bakumenko, O.D.; Zvonarev, A.V.; Kulakovskiy, M.J.; Koleganov, J.F.; Pomerantsev, G.V.; Sergeev, V.A.; Tverdokvsky, N.D.; Savitsky, V.I.; Tsurukin, J.P.; Usanov, V.I. Atomkernenergie; 30: No. 4, 305-307(1977).

From 5. international reactor shielding conference; Knoxville, Tenn. (18 - 22 Apr 1977).

This paper presents the result from experimental and calculational studies carried out at the BN-350 reactor start-up with the aim of determining the efficiency of intra-vessel and biological shielding of the reactor: distributions of reaction rates for various indicators in the upper reactor shield, specific activity values for reactor coolant and gas system, radiation situation within primary circuit cells and in the rooms adjacent to the reactor. The experimental data obtained are compared with the calculated ones.

- 4891 IRRADIATION EFFECTS AND DESIGN OF LMFBR PERMANENT REACTOR STRUCTURES. Blackburn, L.D.; Knecht, R.L. (Westinghouse Hanford Co., Richland, Wash. (USA)). Atomkernenergie; 30: No. 4, 278-281(1977).

From 5. international reactor shielding conference; Knoxville, Tenn. (18 - 22 Apr 1977).

Current approaches to the design of permanent reactor structures for LMFBRs emphasize the restriction of neutron radiation exposure to levels which do not grossly alter the mechanical behavior of the materials of construction. Conventional quantitative design procedures can then be employed to limit calculated stresses to acceptable values. The decrease in material ductility with irradiation has generally been used as a criterion for limiting neutron exposure, and experimental results are examined in the light of establishing such limits. When radiation exposure must exceed the established limits, the properties of irradiated material must be

used to modify design allowable stress intensity values and creep-fatigue interaction. Fracture mechanics analyses are also a potentially important feature of the design procedure in this regime. Irradiation-induced swelling and creep effects also should be accounted for in the design. Several examples of relevant property changes are reviewed.

- 4892 RADIATION PROTECTION/SHIELD DESIGN. A NEED FOR A SYSTEMS APPROACH. Disney, R.K. (Westinghouse Electric Corp., Madison, Pa. (USA)). Atomkernenergie; 30: No. 4, 256-260(1977).

From 5. international reactor shielding conference; Knoxville, Tenn. (18 - 22 Apr 1977).

Radiation protection/shielding design of a nuclear facility requires a coordinated effort of many engineering disciplines to meet the requirements imposed by regulations. In the following discussion, the system approach to Clinch River Breeder Reactor Plant (CRBRP) radiation protection will be described, and the program developed to implement this approach will be defined. In addition, the principal shielding design problems of LMFBR nuclear reactor systems will be discussed in relation to LWR nuclear reactor system shielding designs. The methodology used to analyze these problems in the U.S. LMFBR program, the resultant design solutions, and the experimental verification of these designs and/or methods will be discussed.

- 4893 PREDICTION OF PRIMARY CIRCUIT CONTAMINATION IN POWER REACTORS. Beslu, P. (CEA/CEN de Saclay, France); Costa, L.; Frejaville, G.; Pepin, P. pp 374-391 of Reactor shielding. Roussin, R.W.; Abbott, L.S.; Bartine, D.E. (eds.). Princeton, NJ: Science Press (1977).

From 5. international conference on reactor shielding; Knoxville, TN, USA (18 Apr 1977). See CONF-770401--.

Knowledge of the amount and distribution of activity in the coolant circuit of a power reactor is of great practical importance for determining shielding and for predicting operating personnel radiation exposure. The present status of mathematical models to predict the contamination level in power reactor coolant circuits (High temperature, Fast Breeder and Pressurized Water Reactors) are presented in this paper. Some applications are also given.

- 4894 BN-350 REACTOR SHIELDING STUDIES. Bakumenko, O.D.; Zvonarev, A.V.; Kulakovskii, M.A. pp 494-500 of Reactor shielding. Roussin, R.W.; Abbott, L.S.; Bartine, D.E. (eds.). Princeton, NJ: Science Press (1977).

From 5. international conference on reactor shielding; Knoxville, TN, USA (18 Apr 1977). See CONF-770401--.

This paper presents the results from experimental and calculational studies carried out at the BN-350 reactor start-up with the aim of determining the efficiency of intra-vessel and biological shielding of the reactor: distributions of reaction rates for various indicators in the upper reactor shield, specific activity values for reactor coolant and gas system, radiation situation within primary circuit cells and in the rooms adjacent to the reactor. The experimental data obtained are compared with the calculated ones.

- 4895 BN-350 REACTOR: DEVELOPMENT, DESIGN AND OPERATING EXPERIENCE. Troyanov, M.F.

(Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskij Inst.). pp 775-787 of Nuclear power and its fuel cycle. Vol. I. Vienna; IAEA (1977). (In Russian)

From International conference on nuclear power and its fuel cycles; Salzburg, Austria (2 May 1977).

The author analyses the experience gained with the BR-1, BR-2, BR-5 and BOR-60 reactors, which has served as the basis for the development of the BN-350 reactor with a thermal power of 1,000MW and operating on ^{235}U . The most important design solutions for this reactor are discussed, together with the problems that arose during commissioning. Operating experience since then is described.

- 4896 STUDY OF APPLICABILITY OF AN ALBEDO MONTE CARLO METHOD FOR NEUTRON STREAMING CALCULATION. Kawai, M. (Nippon Atomic Industry Group Co., Ltd., Kawasaki, Japan); Yamauchi, M.; Kadotani, H.; Suzuki, I. pp 636-644 of Reactor shielding. Roussin, R.W.; Abbott, L.S.; Bartine, D.E. (eds.). Princeton, NJ; Science Press (1977).

From 5. international conference on reactor shielding; Knoxville, TN, USA (18 Apr 1977).

See CONF-770401--.

The applicability of albedo Monte Carlo (AMC) method for the calculation of neutrons streaming through cavities and ducts in LMFBR shielding materials was studied. In the present AMC calculation, MORSE-CG code was modified to include the energy-angle dependent differential albedo data, which were generated by ad hoc versions of ANISN code and SLDN code (invariant imbedding method). SLDN code can calculate exactly the albedo data for semi-infinite slab about 100 times as fast as ANISN code. As sample problems, neutron streaming through a sodium straight pipe and a sodium stepped annular pipe penetrating the shielding materials were chosen because the models can be solved exactly by a two-dimensional S_n transport code. It was found that the total neutron fluxes obtained by the AMC calculation agree with those by S_{16} calculation within a statistical error of AMC calculation and that the fast neutron spectrum was affected by the albedo boundary position. As an application, a problem of neutrons streaming through a large void surrounded by concrete was calculated.

- 4897 DEVELOPMENT OF A COMPUTER CODE FOR NEUTRON STREAMING CALCULATION IN LMFBR. Nishimura, T.; Kinjo, H. (Mitsubishi Atomic Power Industries, Inc., Ohmiya, Saitama, Japan). pp 680-687 of Reactor shielding. Roussin, R.W.; Abbott, L.S.; Bartine, D.E. (eds.). Princeton, NJ; Science Press (1977).

From 5. international conference on reactor shielding; Knoxville, TN, USA (18 Apr 1977).

See CONF-770401--.

A practical computer code MADONNA has been developed for neutron streaming calculations in LMFBR shielding with various shapes of void. The code is based upon a modified removal diffusion method coupled with an albedo-transport equation. To facilitate the solution of the albedo-transport equation, the kernel-albedo and the line-of-sight techniques are incorporated. The comparison has been made between this code and a more rigorous discrete ordinates code in a typical reactor shield. The result is quite satisfactory.

- 4898 RESULTS OF NEUTRON PROPAGATION IN STEEL SODIUM MIXTURES WITH VARIOUS SOURCE SPECTRA ON HARMONIE AND TAPIRO. Calamand, D. (CEA, Cadarache, France); Desprets, A.; Rancurel, H.

pp 795-803 of Reactor shielding. Roussin, R.W.; Abbott, L.S.; Bartine, D.E. (eds.). Princeton, NJ; Science Press (1977).

From 5. international conference on reactor shielding; Knoxville, TN, USA (18 Apr 1977).

See CONF-770401--.

The first results of a joint CEA/CNEN neutron propagation program conducted on the source reactors HARMONIE and TAPIRO are presented. In both cases, a buffer zone representative of the blanket of a commercial fast power reactor is interposed between the source reactor and the medium in which neutron propagation is measured. This buffer zone provides a realistic source spectrum to be propagated. Experimental results are compared to older results obtained without the buffer zone. The effect of the source spectrum on neutron propagation is discussed, as well as the coherence of the results obtained with the two installations.

- 4899 INTEGRAL DATA FOR THE PREDICTION OF NEUTRON PROPAGATION IN IRON-SODIUM MIXTURES. Estiot, J.C. (CEA, Cadarache, France); Trapp, J.P.; Palmiotti, G.; Salvatores, M. pp 804-811 of Reactor shielding. Roussin, R.W.; Abbott, L.S.; Bartine, D.E. (eds.). Princeton, NJ; Science Press (1977).

From 5. international conference on reactor shielding; Knoxville, TN, USA (18 Apr 1977).

See CONF-770401--.

This paper presents results of the first stage in the adjustment procedure used to improve the performances of the neutron propagation calculational scheme PROPANE, in a joint effort between CEA and CNEN. For this purpose, integral data for neutron propagation in pure iron and in iron-sodium mixtures are used. The calculational hypothesis and the adjustment procedure are discussed. The performances of this new version of PROPANE are compared to those of the non adjusted one.

- 4900 UK FAST REACTOR SHIELD DESIGN AND PERFORMANCE. Butler, J. (United Kingdom Atomic Energy Authority, Winfrith, Eng.); Brindley, K. pp 819-832 of Reactor shielding. Roussin, R.W.; Abbott, L.S.; Bartine, D.E. (eds.). Princeton, NJ; Science Press (1977).

From 5. international conference on reactor shielding; Knoxville, TN, USA (18 Apr 1977).

See CONF-770401--.

The development of new methods and data-sets for the shield design of CFR1 is described. The calculational routes are being validated using measurements made in mock-ups of large cores on ZEBRA, penetration benchmarks in the ASPIS bulk shield facility, and special thimbles irradiated in the PFR.

- 4901 TWO-DIMENSIONAL SHIELDING BENCHMARKS FOR SODIUM AND IRON AT YAYOI. Oka, Y. (Tokyo Univ.); An, S.; Kasai, S.; Sakata, S.; Miyasaka, S.; Koyama, K.; Hyodo, T. pp 833-840 of Reactor shielding. Roussin, R.W.; Abbott, L.S.; Bartine, D.E. (eds.). Princeton, NJ; Science Press (1977).

From 5. international conference on reactor shielding; Knoxville, TN, USA (18 Apr 1977).

See CONF-770401--.

Two dimensional distributions of neutron flux and gamma ray dose were measured at the fast neutron source reactor 'YAYOI' for two types of shields. One is a 1.8m thick, 1m square sodium shield and the other is a 70m thick, 1m square iron shield. The measurements were performed with threshold detectors, sandwich resonance foils and gold foils for neutrons and thermoluminescence detectors for gamma rays. The results were compared with

calculations by ANISN and TWOTRAN-II. C/E values are not satisfactory on the centerline of the shield. As for the shapes of the distribution of reaction rates in the shields, however, good agreements are obtained between the experiments and the calculations by TWOTRAN.

4902 IRRADIATION PROGRAMME KNK II.

Biermann, W.; Dietz, W.; Duenner, P.; Reichel, H.; Hess, B. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). KFK Nachr.; 9: No. 3, . 64-65(1977). (In German).

The KNK II facility is the first German reactor with a fast core which can be used as a test reactor. The test zone of this reactor consists of plutonium elements of a Mark-Ia-like geometry. Thus a sufficient number of fuel elements with SNR-like geometry and design can be irradiated under SNR-typical operational conditions and investigated with regard to their burn-up. The article presents a survey of the irradiation programme.

4903 FUEL MOTION IN LOSS-OF-FLOW ACCIDENTS.

Lal, D. Atlanta; Georgia Inst. of Tech. (1977). 266p. University Microfilms Order No. 78-02,946.

Thesis (Ph. D.).

Fuel motion is studied in both one [axial] and two [radial and axial] dimensions in the pre-disassembly stage of a loss-of-flow accident in a liquid metal fast breeder reactor. The analysis is based on a model channel, consisting of an isolated single fuel pin with cladding and associated subassembly containment structure. A mathematical model is set up to describe the motion of the seven materials of interest: liquid fuel, liquid steel, solid fuel, solid steel, vapor fuel, vapor steel and gaseous fission products. The multi-field approach is used, avoiding the usual lumping of multiple fluids into a single fluid with average properties. The materials are treated as distinct fields, each with its own equations of density, momentum and energy transport. The various fields are linked together via binary exchange functions for mass, momentum and energy transfer. These exchange functions can be manipulated to give any desired degree of linkage between two fields. The cylindrical region is zoned into a mesh of fixed cells, and the differential equations are approximated by finite difference representations in these cells. The implicitly formulated difference expressions for density and momentum transport are solved iteratively for the density, velocity and pressure distributions. The explicit formulation for energy transport then permits calculation of the internal energy corresponding to these distributions. This numerical scheme has been programmed into the PLOFA computer code, designed for the efficient solution of multi-field confined flow problems. Results for a loss-of-flow accident are presented for single and multi-dimensional calculations. It is observed that an axial analysis is insufficient to predict the phenomena occurring in the initiation phase. The addition of a radial dimension is found to be vitally necessary for realistic accident modeling.

4904 PROGRAMME AND STATUS OF THE DEVELOPMENT OF THE FAST BREEDER REACTOR SYSTEM IN FRANCE.

Vendryes, G. (CEA, 75 - Paris (France)); Besse, G. (Compagnie Generale des Matieres Nucleaires (COGENA), La Bourssidiere, Le Plessis-Robinson, France); Rozenholtz, M. (Novatome, Le Plessis-Robinson, France); Saifcevsy, B. (Societe

Centrale Nucleaire Europeenne a Neutrons Rapides (NERSA), 69 - Lyon (France)). pp 371-390 of Nuclear power and its fuel cycle. Vol. I. Vienna; IAEA (1977). (In French)

From International conference on nuclear power and its fuel cycles; Salzburg, Austria (2 May 1977).

The fundamental justification for fast breeder reactors is based on their breeding ability, and the development of this reactor line is a matter of national importance because there is no reasonable and adequate alternative that could both cover the increasing future energy requirements and provide energy independence for France. The 250MW(e) Phenix power plant has been satisfactorily operated as a demonstration plant for two years now. This proof of the validity of the system was necessary. It underlined its reliability and confirmed the value of the process to the constructors and to the domestic utility, EDF, who confidently decided on industrial and commercial commitment to the new reactor line. This attitude is demonstrated in the following actions in France: (a) the construction by Nersa of the Super-Phenix power plant, a 1200MW(e) prototype commercial power plant which will provide experience in building and operating large fast-neutron power plants; (b) the firm intention of EDF to proceed and bring breeders into operation in the 1990s, representing about 8000MW(e); (c) the creation of a strong industrial organization: the creation in France of the Novatome Company corresponds to this objective - Novatome is now responsible for the industrial development of fast breeder reactors with Creusot-Loire as industrial basis and with all the know-how owned by CEA; (d) the Compagnie generale des matieres nucleaires, Cogema, a CEA subsidiary, is reviewing the actions necessary to reach industrial levels for the cost of the whole fuel cycle. On the other hand, the negotiation of international agreements adds the knowledge and experience gained in the R and D programmes of the various countries concerned. They improve the value and the technical, industrial and commercial basis of this new type of reactor, thus enlarging the prospects of development and commercialization.

4905 BIRTH OF THE BREEDER. Hinton, C. (Electricity Supply Research Council, London (UK)). pp 8-13 of Breeder reactor. Forrest, J.S. (ed.). Edinburgh; Scottish Academic Press (1977).

From Meeting on the breeder reactor; Strathclyde, UK (25 Mar 1977).

The background to the development of the British fast reactor programme is summarized. The problem of design proved to be even more difficult than had been expected and only a small team could be allocated to the work. A zero energy facility was built at Harwell and the small team of engineers at Risley started design of a 60 MW (thermal) experimental reactor. The reactor was commissioned in 1959; it was designed to give experience of control of fast reactors, to give experience of the use of liquid metals as coolants and to provide a facility for the development of fuel elements suitable for industrial fast reactors. It was more than successful in all of these things.

4906 AZIMUTHAL NON-UNIFORMITIES OF TEMPERATURE OF FUEL ELEMENTS ARRANGED IN REGULAR LATTICES FOR TURBULENT LIQUID METAL FLOW. Ushakov, P.A.; Zhukov, A.V.; Matyukhin, N.M. (Gosudarstvennyj Komitet po Ispol'zovaniyu Atomnoj Ehnergii SSSR, Obninsk. Fiziko-Ehnergeticheskiy Inst.). Teplofiz. Vys. Temp.; 15: No. 1, 76-83(1977). (In Russian).

Three methods for semiempirical solution of the problem on nonuniformity of the temperature of fuel elements in regular lattices are considered. An analytical solution based on the hypothesis of analogy of the local heat transfer coefficients with those in equivalent parallel-sided slits is obtained. Another solution is also obtained, with the use of the universal temperature profile in a round pipe. It is believed that the solution in which a temperature profile for a "rod" flow is used for estimating azimuthal, heat flux in the coolant is the principal one. On the basis of an analysis of calculated and experimental data, a simple approximation formula for calculating nonuniformities in fuel element temperature is recommended.

- 4907 VALVES FOR THE SODIUM-COOLED FAST BREEDER KKW KALKAR (SNR 300). Wueliner, G.; Helbig, D. Three R Int.; 16: No. 8, 464-469(1977). (In German).

Valves for controlling the flow of liquid sodium are located in exposed positions. Exacting demands are imposed on the tightness of the pressure vessel and the proper functioning of all components in long-term operation. Valve design was re-examined by stress analysis using finite elements in order to determine what stresses the valves are subjected to as a result of temperature shocks. The bellows valves demonstrated their ability to function properly in prolonged tests under actual service conditions.

- 4908 (EURFNR--1435) FAST BREEDER PROJECT. QUARTERLY REPORT, 1977. Hueper, R. (comp.). (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.)). 1977. Translation by T.J. Clark of KFK--1277/1 176p. Dep. NTIS, PC A09/MF A01.

Work performed within the framework of the Fast Breeder Project at Kernforschungszentrum Karlsruhe during the first quarter of 1977 is described. The work included fuel rod development, materials studies, physics, safety, and sodium technology.

- 4909 DESIGN OF FFTF SHIELDS. Bunch, W.L. (Hanford Engineering Development Lab., Richland, WA). pp 35-43 of Reactor shielding. Roussin, R.W.; Abbott, L.S.; Bartine, D.E. (eds.). Princeton, NJ: Science Press (1977). From 5. international conference on reactor shielding; Knoxville, TN, USA (18 Apr 1977). See CONF-770401--.

The Fast Flux Test Facility (FFTF) is to become the primary test facility in the United States for the development of liquid-metal-cooled, fast breeder reactors. Design philosophy for the facility was to place equipment in shielded cells to maximize the potential for maintenance. Dose rate criteria for each cell were coupled with design-basis radiation source strengths and equipment layouts to establish shielding requirements for each wall throughout the facility. High density concretes were employed to meet physical constraints established by the containment building. Point kernel calculations were made and documented for each wall and floor. Shield acceptance tests are planned to establish dose rates and radiation source strengths for comparison to design calculations.

- 4910 APPLICATION OF ULTRASONIC THERMOMETRY IN LMFBR SAFETY RESEARCH. Carlson, G.A.; Sullivan, W.H.; Plein, H.G. (Sandia Lab, Albuquerque, NM). Ultrason. Symp. Proc.; No. 77CH1264-1, 24-28(1977).

Ultrasonic thermometry has many potential applications in reactor safety research, where extremely high temperatures and lack of visual access may preclude the use of conventional diagnostics. An application (the in-core molten thermometers were used to measure temperatures in UO₂ to incipient melt (2860 C). Each thermometer included five sensor elements 10 mm long, providing five temperatures within the UO₂ at various axial locations. The 10 mm spatial resolution is about five times better than previous applications of the technique. Temperature resolution of $\pm 10^\circ\text{C}$ was indicated by calibration data. Besides providing temperature data approximately 1000°C higher than were obtained with thermocouples, the thermometer yielded valuable axial temperature profile data. 7 refs.

- 4911 SPATIO ENERGETIC SENSITIVITY OF THE EQUIVALENT THERMAL FLUX ON THE INTERMEDIATE HEAT EXCHANGER OF A TYPICAL FAST BREEDER REACTOR. Bololi, A.; Fiorini, L.; Moreau, J.; Suntinger, G. (CEN, Cadarache, France). pp 132-139 of Reactor shielding. Roussin, R.W.; Abbott, L.S.; Bartine, D.E. (eds.). Princeton, NJ: Science Press (1977).

From National Transportation Safety Board meeting; Washington, DC, USA (24 Jan 1977). See CONF-770104--.

The calculation of the thermal equivalent flux on the intermediate heat exchanger is the main problem of a pool type LMFBR Shielding. The present paper is related to a reactor which lateral shielding consists of a sodium-stainless steel mixture. The following problems are treated: Slowing down of neutrons (studied with "Channel Theory"), Sensitivity of the thermal flux on the heat exchanger to the following parameters: stainless steel and sodium total cross section, Legendre polynomial expansion. Some ways for shielding optimizations can be deduced from calculations performed with ANISN and SWANLAKE: use of moderator, local shielding, stainless steel volumetric percentage variable along the lateral shield. Finally, we give an estimate of thickness increase which results from multigroup cross section uncertainties.

- 4912 IRRADIATION EFFECTS AND DESIGN OF LMFBR PERMANENT REACTOR STRUCTURES. Blackburn, L.D.; Knecht, R.L. (Westinghouse Hanford Co., Richland, WA). pp 230-237 of Reactor shielding. Roussin, R.W.; Abbott, L.S.; Bartine, D.E. (eds.). Princeton, NJ: Science Press (1977).

From National Transportation Safety Board meeting; Washington, DC, USA (24 Jan 1977). See CONF-770104--.

Current approaches to the design of permanent reactor structures for LMFBRs emphasize the restriction of neutron radiation exposure to levels which do not grossly alter the mechanical behavior of the materials of construction. Conventional quantitative design procedures can then be employed to limit calculated stresses to acceptable values. The decrease in material ductility with irradiation has generally been used as a criterion for limiting neutron exposure, and experimental results are examined in the light of establishing such limits. When radiation exposure must exceed the established limits, the properties of irradiated material must be used to modify design allowable stress intensity values and creep-fatigue interaction. Fracture mechanics analyses are also a potentially important feature of the design procedure in this regime. Irradiation-induced swelling and creep effects also should be

accounted for in the design. Several examples of relevant property changes are reviewed.

- 4913 NEUTRON TRANSPORT THROUGH STRAIGHT AND ONCE BENT SODIUM PIPE IN SHIELDING OF FAST REACTOR. Burian, J. (Nuclear Research Inst., Rez, Czechoslovakia); Jansky, B.; Osmera, B.; Otopal, P.; Rataj, J.; Petrov, E.; Trykov, L.; Nikolajev, A.; Semenov, V. pp 246-252 of Reactor shielding. Roussin, R.W.; Abbott, L.S.; Bartine, D.E. (eds.). Princeton, NJ; Science Press (1977).

From National Transportation Safety Board meeting; Washington, DC, USA (24 Jan 1977). See CONF-770104--.

The paper presents measurements and calculations on the neutron attenuation along 40 cm diameter straight and bent sodium pipe. The following types of spectrometers were used in measurements on VVR-S reactor: stilbene scintillator, proportional hydrogen counter, multisphere technique and activation foils. Discrete ordinates and Monte Carlo methods were used to study this type of problem.

- 4914 STUDY OF FAST NEUTRON STREAMING THROUGH THE CORE SUPPORT-PLATE SHIELD OF A LMFR. Suzuki, I. (Power Reactor and Nuclear Fuel Development Corp., Tokyo); Oka, Y.; Wakabayashi, H.; An, S.; Sasaki, K.; Miyasaka, S.; Inoue, T. pp 271-279 of Reactor shielding. Roussin, R.W.; Abbott, L.S.; Bartine, D.E. (eds.). Princeton, NJ; Science Press (1977).

From 5. international conference on reactor shielding; Knoxville, TN, USA (18 Apr 1977). See CONF-770401--.

The paper presents measurements and calculations for the fast neutron streaming through the core support plate shield of a LMFR. Experiment was performed in the vertical column of YAXOI, the fast-neutron source reactor at University of Tokyo. The results are arranged in the form of the attenuation factor of the shields which have three types of coolant holes. Two and three dimensional calculations were carried out. The results are compared with experimental ones.

- 4915 ECONOMETRIC ANALYSIS OF ENERGY OVER THE NEXT 75 YEARS. Schmitt, R.W. (General Electric Co., Schenectady, NY); BenDaniel, D.J.; Stewart, P.J.; Manne, A.S. IEEE Trans. Power Appar. Syst.; PAS-96: No. 4, 1353-1361(1977).

From ASME/ASCE joint power generation conference; Buffalo, NY, USA (19 Sep 1976).

An econometric model with a 75-year time-horizon to determine base-load power generation mix is described. A number of conventional, evolutionary and speculative generation systems are considered, together with constraints imposed by fuel availability and technical development times. Preliminary results are presented.

- 4916 STATUS OF PROTOTYPE LARGE BREEDER REACTOR (PLBR) DESIGNS. Minnick, L.E. (Electric Power Research Inst., Palo Alto, CA). Proc. Am. Power Conf.; 39: 152-155(1977). (CONF-770403--).

From 39. annual meeting of the American Power Conference; Chicago, IL, USA (18 Apr 1977).

The background and current developments relating to the PLBR reactor are reviewed.

- 4917 MIXED FUEL CYCLE HYBRID LMFR CORES: DESIGN AND ANALYSIS. Shenoy, S.K. Gainesville, FL; Univ. of Florida (1977).

206p. University Microfilms Order No. 78-10,984.

Thesis (Ph. D.).

The positive sodium-void reactivity effects in large Liquid Metal Cooled Fast Breeder Reactors (LMFRs) may be reduced substantially if the (Pu- ^{239}U) fuel cycle is replaced by the (^{233}U -Th) fuel cycle in the center portion of the cores. Geometric spoiling methods, e.g., reducing the core height-to-diameter ratio, also achieve the same end. Both methods lead to economic penalties in the form of reduced breeding ratio (B.R.) and increased doubling time (D.T.). In this work the relative merits of the two methods of reducing the positive sodium-void reactivity-effect are compared. The calculations show that for comparable improvements in the sodium-void reactivity effect, the economic penalty for using a central core region of (^{233}U - ^{238}U) fuel is smaller than for using a flattened core.

- 4918 HEAT REJECTION FOR THE CLINCH RIVER BREEDER REACTOR PLANT. Donaldson, A.M. (Burns and Roe, Inc., Los Angeles). Proc. Am. Power Conf.; 39: 702-708(1977). (CONF-770403--).

From 39. annual meeting of the American Power Conference; Chicago, IL, USA (18 Apr 1977).

This paper details the optimization and selection of the following cooling systems: mechanical-draft wet tower, mechanical-draft wet/dry tower, hyperbolic or natural-draft tower, and a spray pond.

- 4919 CHEMICAL AND PHYSICAL CHANGES AT SODIUM-STAINLESS STEEL INTERFACES IN FAST REACTORS. Mathews, C.K. (Bhabha Atomic Research Centre, Bombay (India). Radiochemistry Div.). pp 77-90 of Proceedings of the symposium on chemistry and physics of surface of metals and their oxides. Bombay; Department of Atomic Energy (1977).

From Symposium on chemistry and physics of surface of metals and their oxides; Kalpakkam, Tamil Nadu, India (27 Jan 1976).

In the sodium loops of a fast reactor, mass transfer occurs due to the interaction of flowing sodium on stainless steel surfaces. Under the non-isothermal conditions prevailing in the loop some elements are preferentially leached from the surface layers of the hot zone and transported by sodium to the cooled zone where deposition may take place. The available information on the mass transport in non-isothermal sodium loops has been summarised, and an attempt has been made to understand the mechanisms involved, of which the chemical reactions at the sodium-stainless steel interface are especially important. The rate of diffusion towards the solid/liquid interface may be the rate-determining step in some of these reactions. When a ferritic surface layer is formed by the selective removal of austenitic stabilizing elements, diffusion of alloying constituents through the ferritic layer limits the growth of this layer. Only when the surface film is adherent, the diffusion across this layer becomes important. NaCrO_2 , for instance, has poor adherence, and a surface film of this compound may not inhibit further corrosion.

- 4920 FUTURE HELIUM REQUIREMENTS FOR ENERGY RELATED APPLICATIONS. Hilal, M.A.; Stone, E.L.; Van Sciver, S.W.; McIntosh, G.E.; Boom, R.W. Madison, WI; University of Wisconsin (1977). 119p.

Several high technology energy systems under study and development will require large

quantities of helium when used on a commercial scale. The study estimates the helium need for commercial units of these systems, regardless of feasibility, cost or projected use. This report covers fusion reactors, superconductive energy storage, superconducting transmission lines, magnetohydrodynamic (MHD) units, superconducting motors and generators, accelerators, and fission reactors. All these energy systems, except for fission reactors, utilize superconductivity and low temperature helium. Fission reactors may use helium as a gaseous coolant.

- 4921 Industrie nucleaire francaise. (FRENCH NUCLEAR INDUSTRY). Paris, France; CEA (1977). 96p. (In French).

The position of nuclear energy in the world energy market, the nuclear boiler industry (PWR, breeders, small nuclear plants) and the nuclear fuel cycle industry (U ore processing, transformation of U concentrates into hexafluoride, U enrichment, fuel fabrication, zirconium and its alloys, reprocessing) are reviewed. Numerous graphs and tables are given.

- 4922 DOUNREAY PROJECT. Blumfield, C.W. (UKAEA Reactor Group, Dounreay. Dounreay Experimental Reactor Establishment). pp 21-36 of Breeder reactor. Forrest, J.S. (ed.). Edinburgh; Scottish Academic Press (1977). From Meeting on the breeder reactor; Strathclyde, UK (25 Mar 1977).

The paper briefly refers to the history of the Dounreay Experimental Reactor Establishment and the 18 years of safe operation of the Dounreay experimental liquid-metal-cooled fast reactor (DFR). Reference is also made to the successful commissioning of the 250 MW(e) Prototype Fast Reactor (PFR). This long history of the use of fast reactor technology has shown that although fast reactors are different in concept to thermal reactors they are as easily operated, if not more so. The experience gained from the scientific research into materials suitable for fast reactors has provided a comprehensive library of information for the design of the next phase of fast reactors. Fuel assemblies have been operated at full power with flow restricted to produce coolant boiling conditions. These have shown that the high heat fluxes of fast reactors are adequately tamed by the liquid metal coolant. Experience in commissioning the PFR has enabled the designers of the demonstration Commercial Fast Reactor to progressively deal with important issues. In particular, the boiler design, which was always considered to be difficult technological issue, has evolved to eliminate the small water leaks that have occurred in the PFR boilers. This will be tested in the PFR well in advance of its use in the demonstration Commercial Fast Reactor. Reprocessing of the PFR fuel at Dounreay will commence in about a year's time. This has involved reconstructing a plant which was used for reprocessing of DFR fuel. The decommissioning of the plant to the colloquial green field state allowed the modifications to be carried out expeditiously. The new plant has been designed on the knowledge produced in experiments to determine the most efficient and safe method of separation and waste management.

- 4923 Energie nucleaire et energie electrique. (NUCLEAR ENERGY AND ELECTRIC POWER). Boutin, P.; Boutin, P.; Guinier, G.; Mouney, H.; Vincent, F. Paris, France; Eyrolles (1977). 122p. (In French).

The principal scientific aspects of some technical problems posed by nuclear energy

utilization and its transformation into electric power are discussed. Two types of reactors are envisaged in the framework of the present French programs: pressurized water-enriched uranium reactors, and the sodium cooled, fast neutron breeder reactors. After having introduced the concepts of cross sections and radioactive equilibrium, some important questions are outlined: the concept of reactor criticality, the role of delayed neutrons in reactor control, the enrichment of uranium in the 235 isotope. The scientific bases of radioprotection are given: definition of the absorbed dose and dose equivalent; examples of dose determination during an irradiation, of external origin or due to a radioactive contamination. Some aspects including the economy of nuclear power production have been voluntarily rejected, emphasis being put on the bases allowing a fruitful consultation of a more specialized technical and scientific literature in the field.

- 4924 ENERGY GAP AND THE FAST REACTOR. Hill, J. (UKAEA). pp 14-20 of Breeder reactor. Forrest, J.S. (ed.). Edinburgh; Scottish Academic Press (1977).

From Meeting on the breeder reactor; Strathclyde, UK (25 Mar 1977).

The background to the development of fast reactors is summarized. In Britain, the results of the many experiments performed, the operation of the Dounreay Fast Reactor for the past 18 years and the first year's operation of the larger Prototype Fast Reactor have all been very encouraging, in that they demonstrated that the performance corresponded well with predictions, breeding is possible, and the system is exceptionally stable in operation. The next step in fast reactor engineering is to build a full-scale fast reactor power station. There would seem to be little reason to expect more trouble than could reasonably be expected in constructing any large project of this general nature. However, from an engineering point of view continuity of experience is required. If a decision to build a commercial fast reactor were taken today there would be a 14-year gap between strating this and the start of the Prototype Fast Reactor. This is already much too long. From an environmental standpoint we have to demonstrate that we can manufacture and reprocess fast reactor fuel for a substantial programme in a way that does not lead to pollution of the environment, and that plutonium-containing fuel can be transported in the quantities required in safety and in a way that does not attract terrorists or require a private army to ensure its security. Finally, we have to find a way to allow many countries to obtain the energy they need from fast reactors, without leading to the proliferation of nuclear weapons or weapons capability.

- 4925 INTERNATIONAL SCENE. Simpson, H.C. (Strathclyde Univ., Glasgow (UK)). pp 37-48 of Breeder reactor. Forrest, J.S. (ed.). Edinburgh; Scottish Academic Press (1977). From Meeting on the breeder reactor; Strathclyde, UK (25 Mar 1977).

Six main nations or groups of nations are involved in the large scale development of Liquid Breeder Reactors, (1) the U.K., (2) France, (3) West Germany/Belgium/the Netherlands, (4) the U.S.S.R., (5) Japan and (6) the U.S.A. The prototype development programme of the last five of these groups is outlined. France with its 250 MW(e) Phenix plant has had two years of commercial service and has designed a 1200 MW(e) Super-Phenix

plant. Germany, in collaboration with Belgium and the Netherlands, is building a 300 MW(e) plant S.N.R.-300 which is due to go critical by 1981. The U.S.S.R., in addition to the B.N.-350 combined power and desalination plant now in operation, is now building a 600 MW(e) plant B.N.-600 which is due to go critical by the end of 1977, and Japan has also started work on a 300 MW(e) reactor MONJU due to be completed by 1986. Finally, the U.S.A. in addition to a very large experimental programme, is constructing a 380 MW(e) fast reactor at Clinch River, expected to start up in 1982. This implies roughly 2300 MW of electrical power being produced by breeder reactors in the near future.

- 4926 FUEL RECYCLING. Hunt, S.E. (Aston Univ., Birmingham (UK)). pp 53-63 of Breeder reactor. Forrest, J.S. (ed.). Edinburgh; Scottish Academic Press (1977).

From Meeting on the breeder reactor; Strathclyde, UK (25 Mar 1977).

The global nuclear power programme should be designed not only to produce electricity at the lowest possible cost, but also to make the best use of our fissile fuel reserves in the longer term. This clearly indicates the necessity of using breeder reactors, which with plutonium recycling, can achieve total fuel utilisation figures of 70% to 80% as opposed to the very small percentages available from non-breeders, even with recycling. The plutonium can be separated from spent fuel elements chemically. The United Kingdom is in a favourable situation to initiate a fast breeder reactor programme because it has appreciable supplies of plutonium accumulated from the Magnox programme, but on a global scale there is a danger that a sudden expansion of the nuclear programme based on non-breeder reactors will exhaust the supplies of commercially viable uranium before adequate supplies of plutonium have been built up to provide the cores for a significant fast breeder programme. This situation will be worse if, as seems likely, the thermal programmes are based on reactors which are poor producers of plutonium and themselves require enriched fuel. A more modest global expansion of the thermal nuclear programme to about 600,000 MW(e) by the year 2000 is possible using reasonably economic uranium reserves. If this were based on thermal reactors which were reasonably good producers of plutonium (Candu, Magnox and HTR) the programme could provide plutonium for the cores of 800,000 MW(e) of installed fast breeder capacity by the year 2000. Thereafter, if the doubling time of the electricity demand is shorter than that of the plutonium inventory, either the gas-cooled fast breeder or a combination of thermal 'near breeders' and liquid-metal fast breeders will be required.

- 4927 LIQUID METAL FAST BREEDER REACTOR.

Gibson, A.S. (General Electric Co., Sunnyvale, CA). pp 5.102-5.123 of Energy technology handbook. Considine, D.M. (ed.). New York; McGraw-Hill, Inc. (1977).

Information on LMFBR type reactors is presented concerning sodium characteristics, primary coolant circuit, reactor kinetics, design factors and safety considerations, past and present status of LMFBR reactors, fuel cycle considerations, and economic comparisons with other power sources.

- 4928 RADIATION EFFECTS IN BREEDER REACTOR STRUCTURAL MATERIALS. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and

Petroleum Engineers, Inc. (1977). 929p. (CONF-770641--). \$50.00.

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Included are fifty-nine papers. Each paper has been separately processed for the Energy Data Base. (DG)

- 4929 THERMAL AND HYDRAULIC ASPECTS OF NUCLEAR REACTOR SAFETY. VOLUME II. LIQUID METAL FAST BREEDER REACTORS. Jones, O.C. Jr.; Bankoff, S.G. (eds.). New York; American Society of Mechanical Engineers (1977). 336p. (CONF-771120--P2). \$25.00.

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

Separate abstracts were prepared for the individual papers included.

- 4930 EBR-II WIDE-RANGE REACTIVITY METER. Peterson, B.R.; Larson, H.A. pp 28.1-28.25 of Proceedings of the third power plant dynamics, control, and testing symposium. Kerlin, T.W. (ed.). Knoxville, TN; University of Tennessee (1977).

From 3. power plant dynamics, control and testing symposium; Knoxville, TN, USA (7 Sep 1977).

At the Experimental Breeder Reactor II (EBR-II) a wide range reactivity meter has been designed and is being tested. The meter uses flux sensors and a minicomputer system to provide a two-fold function: determination of system reactivity from inverse kinetics calculations and determination of both short-term and long-term anomalous reactivity. A wide range of reactivity measurement is achieved by using an on-line minicomputer to make the calculations over three overlapping ranges. They are the source or counter range, the intermediate range which makes use of Campbell's theorem, and the power or current range. The calculations made in these ranges are taken from proven systems models.

- 4931 FRENCH PROGRAM ON LMFBR CLADDING MATERIALS DEVELOPMENT. Dupouy, J.M. (Centre d'Etudes Nucleaires, Saclay, France). pp 1-11 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

The development of fast breeder reactors is strongly supported in France. The construction of the first reactor of commercial size (1200 MWe) called Super Phenix has been decided and the count-down has already started. As far as metallurgists are concerned, the materials constitutive of the core have already had to be decided upon. The problems at hand to date concern the most correct prediction of burn-up that can be attained and the preparation of the choices for the second core of this reactor and for the first core of the following ones. A brief description is given of research and development on these materials involving various teams in several centers of the CEA, in Cadarache, Grenoble, Marcoule, Fontenay and Saclay.

- 4932 RESEARCH AND DEVELOPMENT PROGRAMME IN THE DEBENE-AREA FOR FAST BREEDER MATERIAL DEVELOPMENT. Dietz, W. (INTERATOM, Bergisch Gladbach, Ger.); Ehrlich, K.; Huet, J.J. pp

13-25 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Since 1964 a joint programme for the development of cladding and core structural materials for Fast Breeder Reactors (FBR) has been underway at the Research Centres in Belgium, Germany and Netherlands and with their industrial partners. Different organizations have contributed to this development in the DEBENE-area [Deutschland (FRG), Belgium, Netherlands], namely the Research Centres: Kernforschungszentrum Karlsruhe and CEN-SCK, Mol, and the industrial partners and Belgonucleaire, Brussel, and Interatom, Bensberg. At the starting point of this development several concepts for FBR's existed, steam cooled, gas cooled, and liquid sodium cooled versions. This was the reason for a relatively broad testing and development programme which took into account the following: (1) commercial austenitic stainless steels with high mechanical strength, (2) commercial nickel-based alloys especially for the purpose of steam cooled FBR's, (3) the development of a series of vanadium-based alloys, and (4) the introduction of dispersion-strengthened ferritic alloys.

4933 UKAEA FAST REACTOR PROJECT RESEARCH AND DEVELOPMENT PROGRAMME ON FUEL ELEMENT CLADDING AND SUB-ASSEMBLY WRAPPER MATERIALS. Harries, D.R. (Atomic Energy Research Establishment, Harwell, Eng.). pp 27-40 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Research and development work on fuel element component (cladding, subassembly wrappers, etc.) materials for the U.K. sodium cooled fast reactor programme has been conducted at the United Kingdom Atomic Energy Authority (UKAEA) establishments at Dounreay, Harwell, Risley, and Springfields during the past fifteen years or so. This work has formed an integral part of, and has been co-ordinated by, the UKAEA Fast Reactor Project and has involved close liaison with the Nuclear Power Company (NPC) and the Central Electricity Generating Board (CEGB). The research and development were initially related to the Prototype Fast Reactor (PFR) but the scope has now been extended to cover the first Civil Fast Reactor (CFR1), which has recently been re-designated the Civil Demonstration Fast Reactor (CDFR). The paper outlines the present status of the development of sodium cooled fast reactors in the U.K. and proceeds to summarize the principal PFR and CDFR core and fuel element parameters which have determined the planning and direction of the fuel element materials programme. The current position on the fuel element cladding and wrapper research and development programme is reviewed, and the facilities and future irradiation programme to be carried out in PFR are described.

4934 EFFECTS OF FAST REACTOR EXPOSURE ON THE MECHANICAL PROPERTIES OF STAINLESS STEELS. Holmes, J.J.; Straalsund, J.L. (Hanford Engineering Development Lab., Richland, WA). pp 53-63 of Radiation effects in breeder

reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

The purpose of the paper is to review the progress made during the last 10 years in understanding the effects of fast reactor exposure on the post-irradiation tensile and creep properties of austenitic stainless steels. The paper emphasizes flow strength levels, stress-to-rupture, and deformation rate of 304 and 316 stainless steels. It is recognized that considerable work has been completed and is in progress on stabilized steels and advanced alloys, but because of the complexity of these systems, it is difficult to arrive at a universally applicable rationalization of response to irradiation. The emphasis on 304 and 316 stainless steels will concentrate on research conducted within the U.S. and U.K.

4935 MECHANICAL PROPERTIES OF IRRADIATED AUSTENITIC STAINLESS STEEL 1.4970. Anderko, K. (Kernforschungszentrum, Karlsruhe, Ger.); Schaefer, L.; Wassilew, C.; Ehrlich, L.; Bergmann, H.J. pp 65-82 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

The paper summarizes the results of post-irradiation tensile and creep tests and of some in-pile stress rupture tests on the stabilized austenitic stainless steel X10NiCrMoTiB 1515 (1.4970). Irradiations were carried out in three reactors: BR2 at Mol (Belgium) up to 1.03×10^{22} n/cm², Rapsodie at Cadarache (France) up to 6.2×10^{22} n/cm² and Dounreay Fast Reactor (Great Britain) up to 7.3×10^{22} n/cm² (all E > 0.1 MeV). Irradiation temperatures were within the range 400 to 720°C. Results for 1.4970 in the reference state: solution annealed + cold worked + aged at 800°C were compared with US data on 20% cold-worked type AISI 316 stainless steel. In addition, the influence of a modified thermal-mechanical treatment on the creep properties of 1.4970 has been investigated. The steel 1.4970 exhibits high yield and creep strengths at testing temperatures above about 600°C due to its high resistance to thermal and irradiation-induced recovery. The biaxial equivalent creep rupture ductility shows a minimum above 0.2%. Furthermore, the influence of the neutron spectrum on the tensile and on the creep rupture ductility of 1.4970 has been studied. TEM investigations were carried out on DFR-irradiated specimens before and after creep testing at 600°C. They showed fast resolution of loops and an annealing of voids accelerated by creep stresses.

4936 POST IRRADIATION MECHANICAL PROPERTIES OF ANNEALED AND COLD WORKED 316 STAINLESS STEEL AFTER IRRADIATION TO HIGH FAST NEUTRON FLUENCES. Dupouy, J.M. (Centre d'Etudes Nucleaires, Saclay, France); Erler, J.; Huillery, R. pp 83-93 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials;

Scottsdale, AZ, USA (19 Jun 1977).

The mechanical properties of the steels used in fast reactor cores are strongly affected by irradiation. The report gives an account of the latest measurements and reports on changes in the properties of the materials irradiated in Phenix.

- 4937 EFFECTS OF IRRADIATION ON THE FRACTURE BEHAVIOR OF AUSTENITIC STAINLESS STEELS. Grossbeck, M.L. (Oak Ridge National Lab., TN); Stiegler, J.O.; Holmes, J.J. pp 95-116 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Fracture in irradiated materials occurs by mechanisms which occur in unirradiated materials in addition to mechanisms related to irradiation phenomena. The paper examines radiation effects in austenitic stainless steels for use as core structural materials in fast breeder reactors.

- 4938 EFFECTS OF IRRADIATION ON FATIGUE AND CRACK PROPAGATION IN AUSTENITIC STAINLESS STEELS. Michel, D.J. (Naval Research Lab., Washington, DC); Korth, G.E. pp 117-137 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

The effects of fast neutron irradiation on low cycle fatigue and fatigue crack propagation in austenitic stainless steels are reviewed. The dependence of fatigue performance on irradiation temperature, neutron fluence, strain range and rate, hold time, and material thermomechanical history is discussed. The effects of neutron irradiation on subcritical crack propagation are reviewed with respect to the influence of irradiation and test temperatures, neutron fluence, hold time effects, and thermomechanical history. Predictive methods for estimating low cycle fatigue and crack propagation performance are discussed.

- 4939 POST-IRRADIATION MECHANICAL PROPERTIES OF AISI TYPE 316 STEEL AND NIMONIC PE16 ALLOY. Barnaby, J. (Dounreay Nuclear Power Development Establishment, Scotland); Barton, P.J.; Boothby, R.M.; Fraser, A.S.; Slattery, G.F. pp 159-175 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

The paper reviews findings of a continuing study of the effects of fast reactor exposure upon the mechanical properties of type 316 steels and Nimonic PE16 Alloy. The aims of this programme are to provide data for design and safety assessment purposes and to further understanding of the relationship between microstructures and deformation and fracture characteristics.

- 4940 EFFECT OF FAST NEUTRON IRRADIATION AND HELIUM INJECTION ON THE MECHANICAL PROPERTIES OF AUSTENITIC STAINLESS STEEL. Koyama, M. (Power Reactor and Nuclear Fuel Development Corp., Ibaraki, Japan); Suzuki, K.; Nakajima, H.; Furuta, T.; Watanabe, R. pp 177-189 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Austenitic stainless steel is a candidate material for the fuel cladding and other core components in the fast breeder reactor. For reactor design and safety analysis it is necessary to define the effects of neutron irradiation on the high temperature mechanical properties of the alloys. It is well established that fast reactor irradiation greatly reduces ductility of austenitic stainless steels. Ductility loss increases monotonically with neutron fluences in both tensile and creep-rupture test. Embrittlement of stainless steel irradiated at below 550°C is due to plastic instability while for the specimen irradiated at above 650°C embrittlement is due to helium produced by transmutation of constituents. The loss of ductility caused in the intermediate temperature range is believed due to both mechanisms. The paper presents the results of a study on mechanical properties of Type 316 stainless steel tube which was irradiated in the Dounreay Fast Reactor at temperature between 250 and 500°C and to fast neutron fluence of 2.0 to 7.8 x 10²² n/cm² (E > 0.1 MeV). The effects of helium injection on the creep rupture properties of Type 316 stainless steel with minor constitution variations as well as the effects of titanium modification on the tensile properties of irradiated Fe - 17Cr - 13Ni base alloys are also presented.

- 4941 IRRADIATION CREEP IN BREEDER REACTOR STRUCTURAL MATERIALS. Straalsund, J.L. (Hanford Engineering Development Lab., Richland, WA). pp 191-207 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Irradiation enhanced creep was first observed in uranium by Roberts and Cottrell in 1955. The observed creep was explained on the basis of the internal stresses developed by the neighboring crystals undergoing anisotropic growth, a mechanism that was expected to operate only in an anisotropic material. Nevertheless, irradiation enhanced tensile stress relaxation in Type 304 stainless steel was observed in 1959 and verified by torsion stress relaxation measurements in 1962. It was not evident, however, whether the creep was a transient or continuing phenomenon until experiments were performed by Lewthwaite et al. in 1967 on several materials in Dounreay Materials Test Reactor. These experiments, which were performed by monitoring the length changes induced in dead weight loaded springs during irradiation, demonstrated the existence of a non-saturating irradiation creep phenomenon for a variety of nominally isotropic materials, including nickel, Nimonic 80-A, and Nimonic PE-16. However, possibly because of the low dose and irradiation temperature, the creep rate of Type 316 stainless steel appeared to

saturate after an initial transient. Experiments of a similar nature were performed by Mosedale and his associates in the Dounreay Fast Reactor in 1969 and continuing creep rates for all of the materials, including Type 316 stainless steel, were observed in this experiment. Thus, the phenomenon of irradiation creep for conditions relevant to fast reactor applications was first reported about two years after the void formation phenomenon was reported by Cawthorne and Fulton in 1967. In contrast to swelling, irradiation creep was considered to have little impact on breeder reactor core performance because of the relatively small creep rates which were observed.

- 4942 IRRADIATION CREEP IN FAST REACTOR CORE COMPONENT MATERIALS. Mosedale, D. (Dounreay Nuclear Power Development Establishment, Thurso, Scotland); Harries, D.R.; Hudson, J.A.; Lewthwaite, G.W.; McElroy, R.J. pp 209-228 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

A technique has been successfully developed for measuring the irradiation creep of metals under tensile stress during 4 MeV proton bombardment. The irradiation creep characteristics of both solution treated and cold-worked AISI Type 321 austenitic steel, as a function of temperature in the range 400° to 600°C and stresses up to 150 MPa at a damage rate of approximately 10^{-6} dpa s⁻¹ and doses up to 0.6 dpa, are described. In addition, specimens have been bombarded with protons under stress at a damage rate of approximately 2×10^{-5} dpa s⁻¹ to approximately 10 dpa in a second apparatus and the irradiation creep rates subsequently measured at the normal 10^{-6} dpa s⁻¹ damage rate, thereby enabling the irradiation creep behavior to be studied in voided material. The data are compared with results obtained by the other workers and related to the existing theoretical models of irradiation creep.

- 4943 RECENT DETERMINATION OF THE IRRADIATION CREEP RATE OF COLD WORKED 316. Dupouy, J.M. (Centre d'Etudes Nucleaires, Saclay, France); Lehmann, J.; Cartier, L.; Huillery, R.; Millet, P. pp 229-240 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Irradiation creep is one of the major consequences of the presence of the high fast neutron fluxes. Its magnitude must be known with accuracy in order to calculate the behavior of the pins, of the wrapper tubes and of the entire core. The most desirable measurement of irradiation creep is naturally the direct, continuous in-pile recording of the deformation of a specimen submitted to a constant stress. This is what is achieved with the in-pile creep measuring device which has been designed, constructed and operated in Rapsodie. The latest measurements are presented and compared with the previous ones and with the published values from other countries.

OF COLD-WORKED W.-NR. 1.4981 IRRADIATED IN THE RAPSODIE REACTOR. Bergmann, H.J. (Internationale Atomreaktorbau GmbH, Bergisch Gladbach, Ger.); Haas, D.; Herschbach, K. pp 241-251 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Biaxially loaded capsules manufactured out of two different heats of W.-Nr. 1.4981, 14% cold-worked, have been irradiated in the French reactor RAPSODIE up to a maximum neutron dose of 50 dpa/sub NRT/ at temperatures ranging from 400 to 505°C. Results of dimensional measurements are reported and discussed with the objective of getting information concerning irradiation induced creep and swelling of the reference steel for the SNR-300 wrapper and structural parts. A comparison of the irradiation creep behavior of W.-Nr. 1.4981 CW with SS 316 CW is made. The swelling of W.-Nr. 1.4981 starts after an incubation time corresponding to a dose of around 20 dpa. A weak temperature dependence between 420 and 480°C, with a maximum at approximately 460°C is noticed. The in-pile creep of W.-Nr. 1.4981 does not show a temperature dependence. Between 15 and 30 dpa, the creep rate increases rapidly. It is thought that the change of the creep rate is related to the onset of swelling. A linear stress dependence of creep is assumed to hold valid although some exceptions at higher dose levels and at 480°C, which are still unexplained, are noticed. Both heats show the same creep and swelling behavior.

- 4945 EFFECTS OF METALLURGICAL VARIABLES ON CREEP ON TYPE 316 STAINLESS STEELS. Kondo, Y. (Sumitomo Metal Industries, Ltd., Amagasaki, Japan); Yukitoshi, T.; Yoshikawa, K.; Nagai, N.; Ohta, S.; Fujiwara, M.; Yoshida, S.; Tanaka, C.; Uematsu, K.; Suzuki, K. pp 253-267 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Creep rupture tests were carried out on non-irradiated and neutron-irradiated type 316 stainless steel cladding tubes. The effects of minor elements, solution treatment temperature, and cold work on the creep rupture properties of the non-irradiated tubes were studied in air and liquid sodium environments. Small additions of phosphorous and boron have remarkable effects on increasing the creep rupture strength due to finely dispersed precipitate inside grains. Cold work increases the rupture strength, but a higher level of cold work beyond a limit has a tendency to reduce the rupture strength at higher temperature and longer duration time due to recrystallization. The strength in sodium at 700°C is lower than in air, accompanied with the significant decarburization. The effects of neutron-irradiation on the creep rupture properties were studied for the sample tubes irradiated at Dounreay Fast Reactor to maximum fluences of about 5.3×10^{22} n/cm² ($E > 0.1$ MeV). The rupture life of the irradiated tubes was about one-tenth that of non-irradiated tubes.

- 4946 IRRADIATION EMBRITTLEMENT OF TYPE 304 STAINLESS STEEL WELDS AND PLATE AT 823 K (550°C). van der Schaaf, B.; de Vries, M.I.;

Elen, J.D. (Netherlands Energy Research Foundation, Petten). pp 307-316 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

The creep and tensile properties of DIN 1.4948, which is equivalent to type 304 stainless steel, have been determined at 823 K after irradiation at 823 K to fast neutron fluences of 10^{23} n . m⁻² and 5×10^{24} n . m⁻² ($E > 0.1$ MeV) and simultaneously to thermal fluences of 6×10^{22} n . m⁻² and 2×10^{24} n . m⁻² respectively. The data are required for the safety analysis of the reactor vessel and inner shields of the loop type sodium-cooled fast breeder reactor, SNR-300. Irradiated specimens from plate material show a creep life reduction of 0.1 compared to the unirradiated condition. Creep strains are reduced from 10% to 15% in unirradiated condition to about 2% after irradiation. Irradiated welded joints show a creep life reduction factor of 0.25; creep strains are decreasing from about 4% in reference condition to 0.5% in irradiated condition. Complementary tensile tests of plate material and welded joints indicate that the observed embrittlement is strain rate dependent; it does not appear at strain rates of 10^{-4} s⁻¹ and higher after a fluence of 10^{23} n . m⁻² and it does not appear for strain rates of 10^{-2} s⁻¹ and higher after a fluence of 5×10^{24} n . m⁻². Additional creep tests with specimens of plate material, irradiated to thermal neutron fluences ranging from 10^{17} n . m⁻² to 10^{22} n . m⁻², show that the thermal "threshold" neutron fluence for the observed embrittlement is 10^{18} n . m⁻², producing an atomic helium fraction of 6×10^{-12} . The irradiation embrittlement is attributed to formation of helium from nuclear reactions of thermal neutrons with the ¹⁰B isotope and its subsequent diffusion to the grain boundaries. Since a drastic reduction of the boron in steels is not feasible, it is suggested to set thermal fluence limits for strain and creep life reduction factors in design criteria.

4947 DUCTILITY OF IRRADIATED TYPE 304 AND 316 STAINLESS STEELS. Blackburn, L.D.; Ward, A.L.; Steichen, J.M. (Westinghouse Hanford Co., Richland, WA). pp 317-326 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Ductility data for irradiated Type 304 and 316 stainless steels were correlated to yield trend curves as a function of temperature and total displacements, a spectrum sensitive exposure parameter. Experimental results are characterized by a linear decrease in ductility with logarithm of displacements (or total fluence) over a substantial range of ductility values at temperatures from 370° to 538°C. The correlation shows that the fluence required to reach a given ductility level (i.e., fluence limit) increases as the neutron energy spectrum becomes softer. Minimum ductility values can be estimated as one-half of nominal values.

4948 EXPLORATION OF THE INFLUENCE OF WELDING VARIABLES ON NOTCH DUCTILITY OF IRRADIATED AUSTENITIC STAINLESS STEEL WELDS. Hawthorne,

J.R.; Watson, H.E. (Naval Research Lab., Washington, DC). pp 327-336 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Postirradiation notch ductility and fracture toughness (K/sub J/) trends of AISI Type 308 weld deposits were explored for radiation exposures in the range of 260 to 649°C. The welds were produced by the shielded metal arc (SMA) process and represented compositional variations (CRE vs non-CRE) and controlled delta ferrite content variations. Fracture toughness determinations were made with fatigue precracked Charpy-V specimens and J-Integral assessment procedures. Specimen irradiations were conducted in EBR-II and UCRR reactors. Large postirradiation decreases in Charpy-V (C/sub V/) energy absorption and fracture toughness were observed. The possibility for K/sub J/ values to be reduced below 88 MPa√m (80 ksi√in.) by moderate fluence exposures was demonstrated. The SMA weld with CRE appeared more sensitive to radiation than a non-CRE Type 308 submerged arc (S/A) weld. An influence of delta ferrite content on 260 or 649°C radiation resistance was not found. Fracture toughness assessments revealed high weld sensitivity to testing rate: a 2 : 1 difference in K/sub J/ level was observed between static vs dynamic test conditions.

4949 COMPARISON OF THE SWELLING PRODUCED IN NICKEL BY NEUTRON IRRADIATION AND BY ION BOMBARDMENT. Smidt, F.A.; Reed, J.R.; Sprague J.A. (Naval Research Lab., Washington, DC). pp 337-346 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Ion bombardment is a useful tool for investigating the swelling behavior of candidate alloys for LMFBR cladding and duct applications because high doses can be attained in a matter of hours. There are, however, several questions which must be answered to increase the utility of the technique. Of major concern is the similarity in void nucleation and growth processes under neutron irradiation and under charged particle irradiation. The difference in damage production rate of some four orders of magnitude produces a change in the temperature dependence of swelling and may also influence the magnitude of the swelling. Differences in the cascade structure of the damage produced by charged particles and neutrons may also influence the results. A few comparisons of swelling produced by charged particles and by neutron irradiation at high doses have been made for technological materials. These results show that the incubation period for swelling to develop and the temperature shift for dose rate are both influenced by alloy composition, heat treatment, and stability. It is therefore desirable to study the phenomena in a pure material so as to separate the effects arising from alloy composition and stability from those inherent to the basic damage process. The swelling behavior of pure nickel has been extensively studied at NRL to characterize the influence of temperature, dose rate, and dose on the nucleation and growth of voids under ion bombardment. Observations are reported of the

swelling behavior of pure nickel of the same stock under neutron irradiation in EBR-II and these observations are compared with the previous ion bombardment results.

- 4950 EFFECTS OF STRAIN RATE ON HIGH TEMPERATURE MECHANICAL PROPERTIES OF IRRADIATED INCOLOY 800 AND HASTELLOY X. Boehm, H.; Closs, K.D. (Kernforschungszentrum, Karlsruhe, Ger.). pp 347-356 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Austenitic steels with about 15% chromium content and about 10% nickel content, respectively, have been mainly used as cladding and core structural materials in the first generation of fast breeder reactors. Previous results on the irradiation behavior of these materials indicate that they probably will not allow attainment of very high burnups. This is due above all to the fact that void formation and irradiation induced creep will result in considerable deformations of the core components and thus set a limit to their service lives in the reactor. For this reason, intensive investigations have been carried on to improve core materials for commercial fast breeder reactors. A promising way seems to be the use of steels having higher nickel contents and of nickel base alloys, respectively. However, the helium production rates of these alloys are higher in the neutron spectrum of fast breeder reactors and thus the problem of high temperature embrittlement might become even more accentuated. Experience on the mechanical behavior under and after irradiation is relatively scarce for these materials. So, in order to get more details about high temperature embrittlement and the irradiation behavior of alloys having high nickel contents, the Incoloy 800 and Hastelloy X materials were irradiated up to high helium concentrations and the dependence of the mechanical properties on the strain rate was studied.

- 4951 RADIATION EFFECTS IN FERRITIC STEELS. Huet, J.J.; Delbrassine, A.; Van Asbroeck, P.; Vandermeulen, W. (C.E.N./S.C.K., Mol, Belgium). pp 357-365 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Ferritic steels were studied in Mol (Belgium) for fast reactor core material applications. For cladding material application, dispersion strengthened ferritic steels were developed. Irradiations were conducted from room temperature to 700°C in the material testing reactor BR2 to investigate the high temperature irradiation embrittlement. In all cases, the ferritic steels were more resistant than the austenitic ones, the difference being very large when the tensile tests were conducted above 550°C. Most of the high fluence fast reactor irradiations are still running. In the few available samples, the density change is small and no single void in the ferritic phase has been detected by transmission electron microscopy. Fuel pins canned with dispersion strengthened ferritic steels were irradiated in the BR2. A burn-up of 120,000 MWd/

t was reached in the oxide pins before a failure occurred despite a very high smeared density (90%), a high temperature (670°C) and a high linear power (560 W/cm). This very encouraging result led to the decision of irradiating ferritic steel canned fuel pins in a fast reactor in realistic conditions.

- 4952 INFLUENCE OF HELIUM ON THE MECHANICAL PROPERTIES OF DIN 1.4970 AUSTENITIC STEEL. Sagases, A.A.; Schroeder, H.; Kesternich W.; Ullmaier, H. (Kernforschungsanlage, Juelich, Ger.). pp 367-375 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Foil specimens of DIN 1.4970 austenitic stainless steel were cyclotron-implanted to uniform He concentrations up to 100 appm. Subsequently, tensile and creep tests at different stress levels and temperatures between 600 and 800°C were performed and compared to the results for He-free control specimens. Whereas the latter exhibit transgranular ductile fracture, the implanted material breaks by intergranular failure with a corresponding reduction of the elongation to fracture values by factors between 3 and 6. SEM and TEM examinations of the microstructure suggest an embrittlement model which is based on the stress-assisted nucleation and growth of helium-filled cavities in the grain boundaries and which correctly describes the observed strong stress and temperature dependence of the rupture time values.

- 4953 THEORY OF PARTICLE REDISTRIBUTION IN AN IRRADIATION ENVIRONMENT. Baron, M.; Chang, A.; Bleiberg, M.L. (Westinghouse Electric Corp., Madison, PA). pp 395-404 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

The stability of strengthening phases in an irradiation environment is an important consideration in the development of alloys for use in LMFBR. It is well known that when excess interstitials and vacancies are introduced by irradiation the diffusion of solute is enhanced. As a consequence, precipitation reactions as well as the redistribution of equilibrium phases can be considerably accelerated. The nucleation and growth of particles of minor phases during irradiation can lead to depletion of swelling inhibitors from solid solution causing excessive accelerated swelling. Furthermore, massive coarsening of precipitates would lead to deterioration of mechanical properties. As an aid for interpretation of experiments it is useful to have a kinetic theory describing the temporal evolution of particle size distributions and related quantities such as particle concentration, average radius and volume fraction of precipitate. The purpose of this investigation is to provide such a theory and to illustrate its usefulness by applying it to the case of in-reactor exposure and ion bombardment of alloys strengthened by γ' precipitates.

4954 COMPARISON OF THE IRRADIATION EFFECTS ON SWELLING AND MICROSTRUCTURE IN COMMERCIAL ALLOY A-286 AND A SIMPLE Fe-25 Ni-15Cr GAMMA PRIME HARDENED ALLOY. Chickering, R.W. (Westinghouse Electric Corp. Madison, PA); Bajaj, R.; Lally, J.S. pp 405-414 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

The irradiation behaviors of alloy A-286 as well as experimental gamma prime hardened alloys are being studied in the National Alloy Development Program for application of gamma prime hardened alloys in the liquid metal fast breeder reactor. The principal direction of the studies concerns the high temperature strength and swelling resistance of the alloys. Minor element compositions may affect the phase stability and void swelling. A high Ti to Al ratio indicates a tendency for the gamma prime $Ni_3(Ti,Al)$ to transform into eta phase (Ni_3Ti) after long term thermal aging and irradiation enhances the tendency for transformation. Another minor element, Si, as a constituent of G-phase, and irradiation may enhance G-phase formation. The Ti, Al, and Si contents affect the swelling of Fe-Cr-Ni alloys. The swelling resistance generally increases with increasing amounts of these three elements in the matrix. In the study the effects of Ti to Al ratio, Ti content, Al content, and Si content on swelling and phase stability were analyzed after Ni-ion irradiation.

4955 INFLUENCE OF FAST DIFFUSING SUBSTITUTIONAL ELEMENTS ON THE SWELLING BEHAVIOUR OF Ni- AND Cu-ALLOYS. Venker, H. (Kernforschungszentrum, Karlsruhe, Ger.); Giesecke, P.; Ehrlich, K. pp 415-420 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Actual theories on swelling operate with mean diffusion coefficients to describe the swelling behavior of alloys. In the model described the partial diffusion coefficients of the components are taken into consideration to predict the swelling behavior under irradiation. From the fact that void formation occurs only if the mobility of the lattice atoms is neither too high nor too low, the conclusion is drawn that in cases where the partial diffusion coefficients of the components of an alloy are widely different, both conditions are less easily fulfilled simultaneously. From this fact a lower degree of swelling can be expected. Simulation experiments with some binary Ni- and Cu-alloys are described. They are in agreement with the model.

4956 EXCESS SUB-SURFACE SWELLING PRODUCED BY NI ION BOMBARDMENT. Johnston, W.G.; Morris, W.G.; Turkalo, A.M. (General Electric Co., Schenectady, NY). pp 421-430 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials;

Scottsdale, AZ, USA (19 Jun 1977).

In an earlier study of the depth distribution of void swelling produced by bombardment with 5 MeV Ni ions at 625°C, anomalously high swelling was observed in the sub-surface region of an annealed Type 304 steel, but not in a Type 321 steel that had been annealed at 1093°C. The latter material is designated as Type 321-B. In both alloys there was a void-free zone of about 50 nm width at the electropolished and ion bombarded surface, and the swelling increased in the second 50 nm, so that at 100 nm and beyond there was no indication of swelling depletion due to loss of point defects to the surface. The term, "sub-surface," is used to denote the region beginning at 100 nm and extending to 200 to 300 nm depth. Additional observations are reported of excess subsurface swelling in an attempt to establish the origin of the effect. As in the earlier work, all of the experiments are with materials containing 15 appm of pre-injected helium.

4957 INFLUENCE OF ALLOYING ON THE IRRADIATION CREEP OF NICKEL. Jones, R.B.; Crossland, I.G. (Berkeley Nuclear Labs., Eng.). pp 443-453 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

In order to investigate the role of alloying on the irradiation creep of nickel, helical springs of three simple representative types of nickel alloy have been exposed under load in the breeder of the Dounreay Fast Reactor. The alloys consisted of a solid solution of nickel containing 14.8 w/o molybdenum, nickel in which about 1 volume % Al_2O_3 particles had been dispersed by powder metallurgy and a conventional age hardening alloy of nickel with 5.9 w/o aluminium heat treated to contain approximately 16 volume % Ni_3Al precipitates (γ'). The springs were stressed in shear between 13 and 28 MPa in a sodium filled rig over the temperature range 280 to 360°C at irradiation doses from 0.1 to 6 dpa. Results up to 4 dpa, from two out of the three irradiation exposures of the rig, have been analyzed. It was found that $Ni-Al_2O_3$ showed slightly less irradiation creep than pure nickel while the irradiation creep of $Ni-Mo$ and $Ni-Al$ was effectively zero except for a small transient component. These results are discussed in relation to the thermal creep behavior of the materials and in terms of published irradiation creep data on springs and pressurized tubes of nickel. It is concluded that alloying additions which lead to thermal creep strengthening also improve the irradiation creep resistance, at least for the relatively low doses encountered in the work described.

4958 IRRADIATION UNDER STRESS AND THE RESULTING MICROSTRUCTURE. Harbottle, J.E. (Berkeley Nuclear Labs., Eng.). pp 455-466 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Internally pressurized, pure nickel tubes covering the stress range 0 to 120 MPa hoop stress have been periodically measured during neutron irradiation at 350°C to investigate the

effect of stress on swelling and its interaction with irradiation creep. At the highest stress the irradiated rupture lifetime only was measured and shown to have increased by a factor of two, corresponding to a strengthening of 20% compared with thermal rupture under identical conditions. The enhancement of swelling by stress, volumetric creep, is found to be a non-monotonic function of stress showing a significant increase in the swelling of 30% at approximately 1 dpa at an intermediate stress of 10 MPa. The increase is transitory, affecting principally the nucleation stage, however, the possibility of a long-term influence on the swelling rate is not rejected. Diameter and length measurements are found to be complementary and lead to a constitutive strain equation. Microstructural examination by TEM shows that the functional dependence of the volumetric creep on stress is followed by the average void volume and that the concentration is the inverse function of this. Final swelling estimated by microscopy agrees with the macroscopic result. The dislocation network after 4 dpa is characterized by bands of high density running in (110) directions irrespective of stress. In addition the tendency for voids to form a lattice is noted. At low doses (0.6 dpa) the rupture specimens exhibit ordered arrays of loops, again (110), and the minimal role of the stress in their formation is postulated.

- 4959 INFLUENCE OF ALLOY CONSTITUTION ON THE SWELLING OF AUSTENITIC STAINLESS STEELS AND NICKEL BASED ALLOYS. Watkin, J.S.; Gittus, J.H.; Standing, J. (UKAEA Reactor Group, Springfields, Eng.). pp 467-477 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

A range of annealed austenitic stainless steels and nickel based alloys have been irradiated unstressed in the Dounreay Fast Reactor to 48 atomic displacements per atom (dpa) at temperatures between 400 and 600°C. During irradiation voids have formed in all materials and caused the specimens, which are right solid cylinders, to swell. The magnitude of swelling has been determined from precise length measurements carried out on the specimens before and after irradiation. From these data the swelling rate with dose has been derived for each alloy at each irradiation temperature which has enabled comparisons of swelling as a function of temperature to be made between the alloys. It is concluded that swelling in Fe-Cr-Ni alloys is substantially controlled by alloy constitution together with a smaller, but not insignificant, contribution from minor elements in solution.

- 4960 VOID SWELLING AND MICROSTRUCTURAL CHANGES IN FUEL PIN CLADDING AND UNSTRESSED SPECIMENS IRRADIATED IN DFR. Bramman, J.I. (Risley Nuclear Power Development Establishment, Eng.); Brown, C.; Watkin, J.S.; Cawthorne, C.; Fulton, E.J.; Barton, P.J.; Little, E.A. pp 479-507 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Structural stability and, in particular, the formation of voids producing swelling under fast neutron irradiation, has been extensively investigated using material irradiated in the Dounreay Fast Reactor as fuel pin cladding and as unstressed cylindrical and discoidal specimens. Fluences up to about 9×10^{22} neutrons cm^{-2} (corresponding to a displacement dose of 40 to 45 displacements per atom NRT) at temperatures covering the range of current design interest have been achieved. Several commercial alloys have been studied in this programme. Among them are unstabilized 316 type stainless steel--in the 20% cold-worked condition, the reference structural material for fuel pin components in PFR--the titanium-stabilized Type 321 (En58B) steel used as wrapper material in PFR, and niobium-stabilized FV548 steel and the nickel based precipitation hardening alloy Nimonic PE16 which are alternative clad and wrapper materials respectively. The early promise of Nimonic PE16 as a low swelling alloy has been confirmed by recent data. For longer term development, ferritic steels have been introduced into the evaluation programme.

- 4961 SWELLING IN NEUTRON-IRRADIATED 300-SERIES STAINLESS STEELS. Appleby, W.K. (General Electric Co., Sunnyvale, CA); Bloom, E.E.; Flinn, J.E.; Garner, F.A. pp 509-527 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Swelling induced by fast neutron irradiation has been characterized in various 300-series stainless steels in the U.S. irradiation program. Data are described which give the temperature dependence of swelling to a fluence of 1.4×10^{23} n/cm², $E > 0.1$ MeV for Types-316 and -304 stainless steels in various metallurgical conditions and to somewhat lower fluences for titanium modified Type-316 and Type-321. Significant differences are found in the swelling response of these materials, particularly with respect to temperature dependencies of swelling and effects of cold working. The development of empirical descriptions of swelling for use in design analyses are described for 20% cold-worked Type-316 and solution-annealed Type-304.

- 4962 VOID FORMATION IN STABILIZED AUSTENITIC STAINLESS STEEL. Ehrlich, K.; Gross, R.; Schneider, W. (Kernforschungszentrum, Karlsruhe, Ger.). pp 529-541 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

The paper summarizes the results of several irradiation experiments in the reactors DFR (Dounreay Fast Reactor) and Rapsodie. Also included are investigations with heavy ions (46 MeV Ni) at the VEC (Variable Energy Cyclotron). The results with respect to the swelling of several niobium- and titanium-stabilized stainless steels DIN 1.4961, 1.4981 and 1.4970 are discussed in terms of minor alloying elements and the type of stabilizer. Also the influence of thermomechanical pretreatments on the swelling threshold and the mean swelling rate are discussed. Finally the dose dependence

of swelling is described for Types 1.4970 and 1.4988 stainless steels.

- 4963 SWELLING BEHAVIOR OF COLD WORKED TYPE 316 STAINLESS STEEL. Uematsu, K. (Power Reactor and Nuclear Fuel Development Corp., Tokyo); Kodama, T.; Ishida, Y.; Suzuki, K.; Koyama, M. pp 571-589 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Dimensional measurement, immersion density measurement and transmission electron microscopy studies have been carried out on one heat of 10% cold worked Type 316 stainless steel claddings irradiated in DFR to fast neutron fluences up to 7.8×10^{22} n/cm² (E > 0.1 MeV), two heats of 10% cold worked Type 316 stainless steel claddings and 10 to 50% cold worked Type 316 stainless steel plate-specimens irradiated in Rapsodie to fast-neutron fluences up to 9.6×10^{22} n/cm² (E > 0.1 MeV). The irradiation induced swelling in Type 316 stainless steel claddings is shown to be strongly dependent upon contents of minor alloying elements such as phosphorus, boron, nitrogen and oxygen. It was observed as irradiation temperature increased that diameter increase, annealing of irradiation hardening and carbide precipitation in 10% cold worked Type 316 stainless steel claddings occurred nearly at the same irradiation temperature. Cold work effectively reduced swelling in Type 316 stainless steel, but the effect was suppressed when polygonization occurred during irradiation.

- 4964 ELECTRON-MICROSCOPY OF FUEL CLADDINGS FROM OCr16Ni15Mo3Nb STAINLESS STEEL IRRADIATED IN THE REACTOR BOR-60 TO BURN-UP 6.12 AND 10.8%. Agapova, N.P.; Ageev, V.S.; Africanov, I.N.; Loboda, E.M.; Unufriev, V.D.; Ostrovsky, Z.E.; Prokhorov, V.I.; Sokursky, Y.N. pp 613-624 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Fuel cladding samples from OCr16Ni15Mo3Nb stainless steel have been examined after irradiation in the Reactor BOR-60 to the fluences 4.2×10^{22} and 6.6×10^{22} n/cm², E > or = 0.1 MeV. Voids, dislocation loops and precipitates are found. It is established that maximum swelling observed near approximately 500°C is equal to 6.5%. Intergranular M₂₃C₆ precipitates and intragranular fine dispersed Nb(C,N) and Laves phase type Fe₂(Mo, Nb) precipitates due to radiation-induced diffusion are found.

- 4965 EFFECTS OF ALLOY COMPOSITION ON VOID SWELLING. Bates, J.F. (Hanford Engineering and Development Lab., Richland, WA); Johnston, W.G. pp 625-644 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Effects of alloy composition on void swelling have been recognized since early investigations of swelling, which indicated that impure metals swell less than pure metals and that certain austenitic stainless steels swell less than others. The sensitivity of swelling to variations in alloy composition offers the possibility of selecting or developing alloys that are more resistant to swelling than those currently in use as fast reactor core structural materials. The discussion presented tends to parallel the several approaches that have been followed in seeking improved alloys: (1) Austenitic stainless steels were originally the preferred core structural materials in the fast breeder reactor programs of various countries. Therefore, an attractive approach is to modify the composition of a commercial austenitic stainless steel to produce a swelling-resistant material while retaining the other desirable properties; (2) The observation that the superalloy, PE16, is much more swelling-resistant than the austenitic stainless steels suggests a second approach, which is to search for existing commercial alloys with suitable properties, including good swelling resistance, for use as a core structural material; (3) Surveys of swelling behavior of a wide range of commercial austenitic alloy by fast neutron irradiation and heavy ion bombardment indicated that resistance to swelling is strongly influenced by the major composition of an alloy. This observed correlation between swelling and Cr and Ni content suggested a third approach, in which the effects of major composition in simple ternary alloys and of minor element additions to such alloys are investigated separately, in order to provide compositional guidelines for selecting or developing swelling-resistant alloys.

- 4966 UKAEA VOID SIMULATION IN FAST REACTOR STRUCTURAL MATERIALS. Makin, M.J.; Hudson, J.A.; Mazey, D.J.; Nelson, R.S.; Walters, G.P.; Williams, T.M. (Atomic Energy Research Establishment, Harwell, Eng.). pp 645-665 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

A general description is given of the two main techniques which have been used in the UKAEA to simulate void swelling, namely 1 MeV electron and high energy ion bombardment, and their characteristics are discussed in relation to the effects produced by neutron irradiation. The experimental results obtained on AISI Type 316, 321 and FV548 austenitic steels, Nimonic PE16 and various other nickel base alloys, and ferritic steels are described in some detail. In particular, the dose and temperature dependencies of swelling and microstructure, the effects of minor compositional variations, heat treatment, cold work, gas content and temperature changes are described. The beneficial effects of minor additions of carbon and silicon are highlighted as well as the general effect of main element composition on swelling characteristics. The similarities and differences in results given by the two techniques are summarized. It is concluded that simulation experiments can provide a useful and reliable guide to the performance of structural materials in a reactor and are particularly useful in the identification of low void swelling commercial alloys.

- 4967 EFFECTS OF IRRADIATION ON SWELLING AND MICROSTRUCTURAL STABILITY OF PRECIPITATION STRENGTHENED ALLOYS. Bleiberg, M.L. (Westinghouse Electric Corp., Madison, PA); Bajaj, R.; Bell, W.L.; Thomas, L.E. pp 667-685 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Neutron and charged particle irradiations were conducted on a series of commercial and experimental alloys covering a wide range of nickel contents and alloy structures. Based upon swelling measurements of aged alloy A-286, a fluence relationship between neutron irradiations, nickel ion bombardment, and electron bombardment was established at the peak swelling temperature. Using this fluence relationship and the calculated alloy matrix compositions, the peak swelling rates of the alloys were calculated based upon nickel ion bombardment results of simple Fe-Ni-Cr ternary alloys. Analysis of the limited neutron data and the extensive nickel ion and electron data indicated that the swelling was controlled by alloy matrix chemistry. However, there was evidence from the electron bombardments that in certain alloys the particles may increase the swelling rate. Experimental and theoretical considerations of the γ' -particle behavior under irradiation in the iron based alloys showed that an equilibrium particle size was obtained in conjunction with an excess of solute in solution over that predicted from thermal kinetics. This implies that overaging cannot occur in these alloys. Also, the excess Ti and Al was beneficial in maintaining the low swelling characteristics of these alloys. Radiation-induced phase instabilities were observed in A-286, In 706, and In 718.

- 4968 INFLUENCE OF METALLURGICAL VARIABLES ON VOID SWELLING IN TYPE 316 STEEL. Terasawa, M. (Toshiba Research and Development Center, Kawasaki, Japan); Shimada, M.; Kakuma, T.; Yukitoshi, T.; Shiraishi, K.; Uematsu, K. pp 687-707 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Void swelling was investigated in a series of modified Type 316 stainless steel alloys that differ in fabrication process and/or chemistry variables. Thirty modified alloys were examined by 200 keV C^+ ion, 5 MeV Ni^+ ion and 1 MeV electron irradiations. Experimental results were presented on the effect of cold working, variations of minor impurity elements, C, P, B and N, and addition of Ti or Nb to 20% cold worked Type 316 steel, and heat treatments. General trends of swelling behavior were established for most of the experimental variables, though some exceptions existed. Of these results, Ti addition was most stimulating. The alloys with Ti showed no swelling when solution treated, but rather high swelling when cold worked by 10 to 20% or aged by certain thermo-mechanical treatment. Considering that the swelling inhibition was closely related to C content and the trend was also recognized for Nb addition, it may be concluded that Ti, Nb or C only in solutions is effective in reducing swelling and that any precipitation enhances swelling.

- 4969 EFFECT OF STRUCTURE AND ALLOYING ELEMENTS ON VOID FORMATION IN AUSTENITIC STEELS AND NICKEL ALLOYS. Levy, V.; Azam, N.; Le Naour, L.; Didout, G.; Delaplace, J. (Centre d'Etudes Nucleaires, Saclay, France). pp 709-725 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

One of the major problems in the development of a fast breeder reactor is the phenomenon of metal swelling due to the formation of radiation induced voids. In this field we are faced with two problems: provide estimates of the swelling that must be taken into account in the design of power reactors and develop alloys with very low swelling rates. To cope with these problems on a reasonable time scale a great deal of work has been developed using high flux ion or electron beams to reach the displacement dose levels involved in a power reactor. However the use of either tool for rapid simulation of the fast neutrons environment requires the knowledge of experimental and theoretical correlations between the swelling caused by neutrons and the swelling caused by other particles. In the case of pure metals a thorough investigation of the mechanisms involved in void swelling has led the Physical Metallurgy research group of the Commissariat a l'Energie Atomique to elaborate a simulation model which reproduces reasonably well the experimental swelling versus temperature curves for all the irradiating particles if dislocation densities are available.

- 4970 RADIATION DAMAGES IN CONSTRUCTIONAL STEELS AND HIGH-NICKEL ALLOYS IRRADIATED WITH NEUTRONS. Scherbak, V.I.; Bykov, V.N.; Vorobiev, A.N.; Dmitriev, V.D. (Institute of Physics and Power Engineering, Obninsk, USSR). pp 773-779 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Results of electron microscope investigations of IKH12M2S2, OKH18N9T and OKH16N15M3B steels and OKH16N40B and OKH16N80B alloys irradiated at 500°C at doses up to 2.5×10^{22} n/cm² are presented. The dependences on nickel content of voids, dislocation loops and precipitate concentrations and mean size are discussed. The different viewpoints on the influence of the chemical composition on void swelling and dislocation structure development are considered. From experimental results it follows that the chemical composition change favorably affects the dislocation capability to capture point defects.

- 4971 IRRADIATION CREEP-SWELLING COUPLING: MICROSTRUCTURAL MODELING AND DATA ANALYSIS. Weiner, R.A.; Foster, J.P.; Boltax, A. (Westinghouse Advanced Reactors Div., Madison, PA). pp 865-878 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials;

Scottsdale, AZ, USA (19 Jun 1977).

The irradiation creep behavior predicted by a model of microstructural evolution is compared with irradiation creep data on solution-annealed Type 304L stainless steel. These data are shown to be consistent with an irradiation creep-swelling coupling in which the swelling enhanced irradiation creep rate is proportional to the swelling raised to a power. The implications of this functional form for the irradiation creep-swelling coupling on the irradiation creep behavior of reactor components with time dependent stress and temperature histories, including the results to be expected from a comprehensive reactor history effects experiment, are discussed. The differences in the irradiation creep behavior of solution-annealed and cold-worked steels are investigated theoretically by incorporating the dose and temperature dependence of the line dislocation density into the microstructural model. This refinement of the microstructural model predicts that the ratio of the irradiation creep in cold-worked and solution-annealed steels will be dose dependent, the irradiation creep material constants will be temperature dependent, and below 550°C swelling enhanced irradiation creep will be the same in cold-worked and solution-annealed Type 316 stainless steel.

- 4972 COMPUTER SIMULATION AND NEUTRON, HEAVY-ION AND ELECTRON IRRADIATION CORRELATION. Ishino, S.; Iwata, S.; Matsutani, Y.; Tanaka, S. (Japan Atomic Energy Research Inst., Ibaraki, Japan). pp 879-898 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

Detailed calculations of various damage parameters as displacement cross section, PKA energy spectrum etc. have been performed for Fe, Cr, Ni and standard stainless steel for the neutron spectrum of the core centre of MONJU. The calculation takes into account elastic scattering as well as various non-elastic nuclear reactions as (n,n') , $(n,2n)$, (n,p) and (n,α) , which become more important for higher neutron energies.

- 4973 FUNDAMENTAL ASPECTS OF THE EVOLUTION OF, AND PHASE CHANGES IN, METALS AND ALLOYS UNDER IRRADIATION. Martin, G.; Bocquet, J.L.; Barbu, A.; Adda Y. (Centre d'Etudes Nucleaires de Saclay, Gif-sur-Yvette, France). pp 899-915 of Radiation effects in breeder reactor structural materials. Bleiberg, M.L.; Bennett, J.W. (eds.). New York; American Institute of Mining, Metallurgical, and Petroleum Engineers, Inc. (1977).

From Conference on radiation effects in breeder reactor structural materials; Scottsdale, AZ, USA (19 Jun 1977).

The quantitative prediction and the simulation of irradiation damage present a considerable interest for the technology of fast neutron and fusion reactors. However, at present time, even qualitative predictions are not always possible when the dose rate varies over a relatively large range; for example the formation of vacancy loops takes place instead of that of voids, precipitation occurs in under-saturated solid solutions etc. Therefore, in addition to a microscopic description of the phenomena, it seems important to elaborate an irradiation damage phenomenology. The results are reported of a systematic experimental study

of dose rate effects for radiation induced precipitation in Ni Si solid solution alloys. As will be seen, the existence of dose rate thresholds for this phenomenon is clearly established. The composition dependence of this threshold permits the definition of a solubility limit under irradiation which is both temperature and dose rate dependent. Finally, the various theoretical approaches to radiation induced precipitation are discussed.

- 4974 NATURAL CIRCULATION HEAT TRANSFER IN POOL-TYPE LMFBR'S. Singer, R.M. (Argonne National Lab., IL); Grand, D.; Martin, R. pp 239-264 of Thermal and hydraulic aspects of nuclear reactor safety. Vol. II. Jones, O.C. Jr.; Bankoff, S.G. (eds.). New York; American Society of Mechanical Engineers (1977).

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

The authors present a state-of-the-art review of natural convection heat transfer in pool-type LMFBR's. Studies related to thermal stratification of the sodium in the primary vessel are reported and include topics such as mixed convection in vertical plane channels and rectangular cavities, and thermal interactions between the hot and cold regions of two-zoned reactor vessels. Results from laboratory and plant tests concerned with the natural circulation behavior of reactor primary heat transport systems are also summarized. From this review it is apparent that much is presently known about steady-state natural convection behavior of LMFBR's, but the study of transient phenomena is not well understood. 40 references.

- 4975 THERMAL HYDRAULIC RESPONSE OF AN LMFBR PLANT TO UPSET EVENTS. Bein, M.; Yahalom, R. (General Electric Co., Sunnyvale, CA). pp 8.1-8.17 of Proceedings of the third power plant dynamics, control, and testing symposium. Kerlin, T.W. (ed.). Knoxville, TN; University of Tennessee (1977).

From 3. power plant dynamics, control and testing symposium; Knoxville, TN, USA (7 Sep 1977).

The paper describes briefly a computer program for analyzing LMFBR plant dynamics and thermal hydraulic response of the system to some typical upset events. The reference plant design is a commercial size 900 MWe plant with a saturated steam cycle. The response of this system to a normal scram is compared with that of a superheated steam cycle. It is expected that the saturated steam cycle with the inherent heat capacity will have a better chance to sustain severe temperature transient than a design with less water inventory.

- 4976 HYBRID COMPUTER MODEL OF THE PROTOTYPE LARGE BREEDER REACTOR PLANT AND CONTROL SYSTEM. Walsh, J.M.; Kesavan, K. (Westinghouse Electric Co., Madison, PA). pp 20.1-20.26 of Proceedings of the third power plant dynamics, control, and testing symposium. Kerlin, T.W. (ed.). Knoxville, TN; University of Tennessee (1977).

From 3. power plant dynamics, control and testing symposium; Knoxville, TN, USA (7 Sep 1977).

The objective of the Hybrid Computer Model is to provide an analytical method for design and analysis of the Prototype Large Breeder Reactor (PLBR) plant and reactor control systems. The PLBR is a 3-loop, commercial-size plant. The Model is arranged on a two-loop basis, with one of the loops representing one

plant loop and the other representing the remaining plant loops lumped. The control systems for the Nuclear Steam Supply System (NSSS) are simulated on the analog computer with the exception of the control rod drive mechanism (CRDM) and controller which are simulated on the digital computer. The reactor, sodium pump drives, and reactor power setpoint generation are also included in the analog computer portion of the PLBR Hybrid Computer Model. The digital computer portion of this hybrid model includes the primary and intermediate sodium pumps and hydraulics, intermediate heat exchanger, steam generator system, and steam turbine system and controls. Results of plant and reactor control system analysis are provided including plant power transients and control system step responses at design and part load conditions. Plant power transients from 100% to 60% and back to 100% power conditions at $\pm$ 3% per minute ramp rates have been obtained and demonstrate high performance operation for the total PLBR plant and controls. Plant power demand steps of $\pm$ 10% rated power have demonstrated a high performance, stable, and coordinated plant control system over an operating range of 100% to 40% of rated power conditions. The results of the analysis show that the feed-forward and feedback NSSS control systems operate in a coordinated fashion to maintain design steam temperature and pressure conditions at the turbine throttle under plant power transient conditions.

- 4977 NATURAL CIRCULATION IN FFTF, A LOOP TYPE LMFBR. Additon, S.L.; Parziale, E.A. (Hanford Engineering Development Lab., Richland, WA). pp 265-283 of Thermal and hydraulic aspects of nuclear reactor safety. Vol II. Jones, D.C. Jr.; Bankoff, S.G. (eds.). New York; American Society of Mechanical Engineers (1977).

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

The authors present a state-of-the-art review of natural circulation heat transfer in loop type reactor plants. Most of the examples are taken from Fast Flux Test Facility (FFTF) design experience, drawing on the authors' familiarity and a developing base of available documentation. On-going studies related to the Clinch River Breeder Reactor (CRBR) and some foreign experience are also noted where available in the literature. The emphasis is on the role of natural circulation in decay heat removal; however, free convection during either operation at power or normal shutdown does influence some aspects of the design and these are reviewed. In treating decay heat removal the topics discussed include steady state loop performance and transient dynamics for conditions immediately after scram and for the longer term which involves different considerations. The review summarizes complex dynamics, specific to the FFTF design evaluation, which particularly illustrate large available margins for decay heat removal (possibly involving coolant boiling). Also addressed are modelling considerations and the status of experimental programs for loop reactor applications. 28 references.

- 4978 STATUS AND SUMMARY OF TREAT IN-PILE EXPERIMENTS ON LMFBR RESPONSE TO HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS. Dickerman, C.E.; Rothman, A.B.; Klickman, A.E.; Spencer, B.W.; De Volpi, A. (Argonne National Lab., IL). pp 19-50 of Thermal and hydraulic aspects of nuclear reactor safety. Vol. II. Jones, D.C. Jr.; Bankoff, S.G. (eds.). New York;

American Society of Mechanical Engineers (1977).

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

A program of transient in-reactor experiments is being conducted by Argonne National Laboratory in the Transient Reactor Test (TREAT) facility to guide and support analyses of hypothetical core disruptive accidents (HCDA) in liquid metal fast breeder reactors (LMFBR). Test results provide data needed to establish the response of LMFBR cores to hypothetical accidents producing fuel failure, coolant boiling, and the movement of coolant, molten fuel, and molten cladding. These data include margins to fuel failure, and modes of failure and movements, and evidence for identification of the mechanisms which determine the failure and movements. A key element in the program is the fast neutron hodoscope, which detects fuel movement during experiments as a function of time within the test apparatus. 47 references.

- 4979 SODIUM BOILING IN FAST REACTORS: A STATE-OF-THE ART REVIEW. Peppler, W. (Gesellschaft fuer Kernforschung m.b.H., Ger.). pp 123-153 of Thermal and hydraulic aspects of nuclear reactor safety. Vol. II. Jones, D.C. Jr.; Bankoff, S.G. (eds.). New York; American Society of Mechanical Engineers (1977).

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

Because of the importance regarding severe hypothetical accidents in liquid metal fast breeder reactors, sodium boiling has been studied in depth, including conditions of natural convection. This paper summarizes the work accomplished and delineates the need for future work in areas of loss of flow, local cooling disturbances, and decay heat removal. 46 references.

- 4980 CONTRIBUTION TO THE STUDY OF SODIUM BOILING DURING SLOW PUMP COASTDOWN IN LMFBR SUBASSEMBLIES. Costa, J. (Centre d'Etudes Nucleaires de Grenoble, France). pp 155-169 of Thermal and hydraulic aspects of nuclear reactor safety. Vol. II. Jones, D.C. Jr.; Bankoff, S.G. (eds.). New York; American Society of Mechanical Engineers (1977).

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

Sodium boiling phenomena have been studied both experimentally and analytically within the framework of a hypothetical slow pump coastdown without scram accident for a large fast breeder reactor of the Super Phenix type. The voiding process of the channel is governed by a two phase flow instability called a flow excursion. Boiling in the single channel approximation of a subassembly is described. Considerations of the bundle effects, based on experiments in a 19 pin bundle, are also given. 46 references.

- 4981 MELTING, BOILING AND FREEZING: THE "TRANSITION PHASE" IN FAST REACTOR SAFETY ANALYSIS. Epstein, M. (Argonne National Lab., IL). pp 171-193 of Thermal and hydraulic aspects of nuclear reactor safety. Vol. II. Jones, D.C. Jr.; Bankoff, S.G. (eds.). New York; American Society of Mechanical Engineers (1977).

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

Information regarding the transition phase

of a hypothetical accident sequence in a liquid metal fast breeder reactor is summarized. The review covers the current work on phenomenological sequences, molten clad behavior preceding and during fuel melting, boiling regimes with internal heat generation, and freezing of molten fuel-steel mixtures. 51 references.

- 4982 FUEL COOLANT INTERACTION IN FAST REACTORS. Board, S.J. (C.E.G.B. Berkeley Nuclear Labs., Eng.); Calderola, L. pp 195-222 of Thermal and hydraulic aspects of nuclear reactor safety. Vol. II. Jones, O.C. Jr.; Bankoff, S.G. (eds.). New York; American Society of Mechanical Engineers (1977).

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

After a brief historical review, the most significant experiments are described and their results discussed and compared. The experiments range from basic studies with nonreactor materials to impile fuel failure tests. In all those involving fast reactor materials, only low efficiency interaction has occurred, apart from the case of sodium injection into fuel. The main theoretical models are critically reviewed in the light of the experimental evidence. Three main stages in energetic FCIs have been identified: an initial quiescent period in which the two fluids intermix and which requires film boiling; a small disturbance inducing local interaction; and a coherent propagation through the interacting masses. It is not yet possible to define completely the conditions under which these stages can occur, and there is not yet enough evidence to be completely confident that this description applies to the injection experiments of sodium into UO_2 . 104 references.

- 4983 EVALUATION OF THE POTENTIAL FOR ENERGETIC FUEL-COOLANT INTERACTIONS IN HYPOTHETICAL LMFBR ACCIDENTS. Henry, R.E.; Cho, D.H. (Argonne National Lab., IL). pp 223-237 of Thermal and hydraulic aspects of nuclear reactor safety. Vol. II. Jones, O.C. Jr.; Bankoff, S.G. (eds.). New York; American Society of Mechanical Engineers (1977).

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

Mixing of hot and cold liquids can produce an explosive event when the temperature of the hot liquid is much higher than the boiling point of the cold liquid. The sudden vaporization of the cold liquid has been identified as a likely cause of this vapor explosion, currently termed a fuel-coolant interaction within the nuclear community. Size and time scales of such interactions are such that current understanding of detailed physical mechanisms is limited. Two important aspects of this field of research are discussed including nucleation and mixing, and the conclusion is drawn that large scale vapor explosions are unlikely in liquid metal fast breeder reactors having oxide fuels, although additional work in the area of rapid mixing during short period overpower transients is required. 35 references.

- 4984 RESIDUAL STRESS MEASUREMENT IN AUSTENITIC ALLOY TUBES. Gelpi, A.; Archer, J. (Novatome Industries, 92 - Le Plessis-Robinson (France)); Donati, J.R.; de Keroulas, F. (Electricite de France, 77 - Evreux). pp 332-337 of 4. Colloque international sur les methodes analytiques par rayonnements X.

Strasbourg, 9-13 mai 1977. Issy-les-Moulineaux, France; Compagnie Generale de Radiologie (1977). (In French)

From 4. international colloquium on x-ray analytical methods; Strasbourg, France (9 May 1977).

Residual stress measurement methods are presented and compared: stress gages and longitudinal cutting, stress corrosion test, X-ray diffractometry. Examples on 600 and 800 alloys and 18-10 stainless steel employed in nuclear station steam generator tubes are given.

- 4985 ON THE OPTIMIZATION OF A FAST HIGH-POWER REACTOR OF LMFBR TYPE BY MEANS OF A NON-CONVEX PROGRAMMING MATHEMATICAL MODEL. Pavelescu, M.; Dumitrescu, H.; Ochiana, G. (Inst. of Nuclear Power Reactors, Bucharest, Romania). Rev. Roum. Phys.; 22: No. 9, 997-1003(1977).

The authors consider the problem of minimizing the doubling time for an LMFBR. The decision vector is a distribution one of fissile and fertile material in the reactor which is considered a static system in initial state, and the mathematical model for optimization is a non-convex programming model. Modeling of the optimization problem is carried out by the SUMT method. 4 refs.

- 4986 PRODUCTION AND EMISSION OF CARBON-14 FROM NUCLEAR POWER STATIONS AND REPROCESSING PLANTS AND ITS RADIOECOLOGICAL SIGNIFICANCE. Bonka, H.; Bruessermann, K.; Schwarz, G.; Willrodt, U. (Technische Hochschule Aachen (Germany, F.R.)). pp 945-948 of Proceedings of the 4. International congress of the International Radiation Protection Association. Vol. 3. Fontenay-aux-Roses, France; Association Internationale de Protection contre les Rayonnements (1977).

From 4. congress of the International Radiation Protection Agency; Paris, France (24 Apr 1977).

In nuclear power reactors with 1000 MWe full load capacity Ci^{14} will be produced at rates of 10 to 100 Ci/a according to the reactor type. The yearly emission rate for LWR is assessed at 10 Ci, for HTR at 0.1 Ci and for LMFBR at 1 Ci. The corresponding RP for 40,000 MWe installed nuclear reactor capacity may release about 500, 3000 and 100 Ci/a according to the nitrogen impurities in the fuel elements and the coolant of the reactor type. In view of these assumptions the total body dose of the population in the vicinity of present commercial power reactors in the Federal Republic of Germany is significantly influenced by Ci^{14} , although the entire dose itself is very low. Regarding the collective dose to the world population radiocarbon is of the greatest importance for both nuclear power plant emissions and for gaseous wastes from reprocessing plants because of its wide dispersion in the ecosystems. The exposure due to globally distributed Ci^{14} is expected to increase significantly from the year 2000 on. In the long term it is therefore necessary to find a way of lowering the emissions, for example by reduction of the N-impurities, or by collection of radiocarbon with krypton control systems in RP, or by changing the head-end process of HTR-RP.

- 4987 POST-ACCIDENT HEAT REMOVAL IN LMFBR'S. Glueckler, E.L. (General Electric Co., Sunnyvale, CA); Baker, L. Jr. pp 285-324 of Symposium on the thermal and hydraulic aspects of nuclear reactor safety. Vol. 2. Jones, O.C. Jr.; Bankoff, S.G. (eds.). New York; American Society of Mechanical Engineers

(1977).

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

For safety assessments of early sodium cooled fast breeder reactors (LMFBRs), extremely unlikely events such as unprotected loss of flow and transient overpower sequences have been evaluated. The prediction of the consequences of these postulated events involves investigations of decay heat dissipation and core retention. In general, these problem areas are referred to collectively as post-accident heat removal (PAHR). This paper reviews data on heat transfer, fluid mechanics, nuclear stability and material interactions associated with PAHR. Some concepts applied in early LMFBRs for consequence mitigation are also described.

- 4988 TIME-RESOLVED CODED APERTURE IMAGING SYSTEM FOR NUCLEAR FUEL MOTION MEASUREMENTS. Stalker, K.T.; Kelly, J.G. (Sandia Labs., Albuquerque, NM). pp 63-70 of Proceedings of the Society of Photo-Optical Instrumentation Engineers. Bellingham, WA; Society of Photo-Optical Instrumentation Engineers (1977).

A coded aperture imaging system for the measurement of nuclear fuel motion in simulated core disruptive accidents is described. A one dimensional Fresnel zone plate type coded aperture is used to modulate the fission gamma rays emitted from the fuel pin under test. Two active imaging systems have been developed to detect the coded image and record it on film using a high speed camera. The first system is an X-ray image intensifier which converts fission gamma rays to optical photons in a thin (0.1 mm) CsI(Tl) scintillator screen on which a photocathode is deposited. The photoelectrons thus created are collected and focused by electron optics and produce an image on the output phosphor. The second system utilizes a NaI(Tl) scintillator crystal and a separate high gain optical image intensifier to produce the desired coded image. With a few modifications both systems appear capable of achieving the preliminary design goals of 1 mm radial resolution, 5 mm axial (along the fuel pin axis) resolution and 1 msec temporal resolution. Presently, the X-ray image intensifier system offers the best resolution, but the second system provides higher gain and a wider selection of operating conditions. The temporal response of both systems is limited to a few milliseconds by the output phosphor. Experimental performance data for each system will be presented, and the results of preliminary nuclear fuel pin imaging experiment performed at the Sandia Pulsed Reactor will be shown.

- 4989 NUCLEAR REACTOR SAFETY HEAT TRANSFER. Bishop, A.A.; Kulacki, F.A.; Bankoff, S.G.; Jones, D.C. Jr. (eds.). New York; American Society of Mechanical Engineers (1977). 66p. (CONF-771120--P4).

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

Included are seven papers, each of which has been separately processed for the Energy Data Base. (DG)

- 4990 ELECTROMAGNETIC FLOWMETERS FOR FAST REACTORS. Baker, R.C. (Imp Coll of Sci and Technol, London, Engl). Prog. Nucl. Energy; 1: No. 1, 41-61(1977).

This paper is in four parts. Section 1 explains the theory of the induced-voltage electromagnetic flowmeter and then considers

various types which have been used. In Section 2 flux-distortion flowmeters are described. These have been proposed mainly for monitoring the primary circuit flow and again both flow-through and probe types have been tested. Sections 3 and 4 continue the discussion of the flux-distortion flowmeter by introducing two methods of analyzing its performance. In Part 4 the second linearized method uses current rings to divide up the conducting material. By considering the interaction of all the rings, it is possible to obtain the current distribution and hence the magnetic field. In conclusion it is suggested that further development would be useful of the devices which are most suited to the liquid metal fast breeder reactor. 58 refs.

- 4991 HEAT TRANSFER, SODIUM BOILING AND FUEL PIN INTEGRITY AFTER A SEVERE LMFBR PIPING FAILURE. Gartside, C.H.; Hwang, I.T.; Marr, W.W.; Miles, K.J.; Tessler, J.H.; Thompson, D.H. (Argonne National Lab., IL). pp 9-15 of Nuclear reactor safety heat transfer. Bishop, A.A.; Kulacki, F.A.; Bankoff, S.G.; Jones, D.C. Jr. (eds.). New York; American Society of Mechanical Engineers (1977).

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

A severe LMFBR piping system failure has been simulated with a thermal-hydraulically prototypic in-reactor experiment. The consequences of the simulated accident on the test fuel were much less than anticipated. Sodium boiling was stable and well behaved and no clad or fuel damage was observed. Analysis of the test data with the current heat transfer and sodium boiling models has demonstrated their conservatism. Sodium boiling is not well described by a single channel slug expulsion model and the two dimensional effects of coolant mixing and heat conduction may have an important impact on the boiling process. The predicted fuel-cladding gap conductance may be too small.

- 4992 HEAT AND MASS TRANSPORT PROCESSES OF A HORIZONTAL MELTING OR DECOMPOSING LAYER UNDER A MOLTEN POOL. Alsmeyer, H.; Reimann, M. (Kernforschungszentrum, Karlsruhe, Ger.). pp 47-53 of Nuclear reactor safety heat transfer. Bishop, A.A.; Kulacki, F.A.; Bankoff, S.G.; Jones, D.C. Jr. (eds.). New York; American Society of Mechanical Engineers (1977).

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

After a hypothetical core disruptive accident of an LWR or FBR, melting of solid boundaries in contact with molten fuel may occur. Some of the melting materials of the structures or sacrificial barriers may be lighter and immiscible with the fuel. The influence of the Taylor-Helmholtz instability on the melting of horizontal plates under fuel has been investigated in different model experiments for the situation given above. A very regular square grid, as predicted from stability analysis, was found to govern the flow and heat transfer mechanisms for melting materials of large and small density differences. When melting is combined with the formation of crusts or evaporation of the molten material, Taylor instability again has an important effect. A two-component melting and evaporating horizontal plate was used to study the thermohydraulic aspects of decomposing concrete beneath UO_2 . The escaping gas flow was found to dominate the transport processes and to be similar to the Taylor instability flow. A theoretical model is

derived for the decomposition of concrete under the influence of a high heat flux. The decomposition process consists of a melting process at the surface of the material and of gas/steam release which will occur at well-defined temperatures in the depth of the material. The formation of a gas/steam layer on the surface of the concrete and the formation and departure of bubbles from this layer is very similar to film boiling on horizontal surfaces. Therefore, a hydrodynamical model is derived corresponding very closely to Berenson's film boiling model. The heat transfer between the bulk liquid and the decomposing material through the gas/steam layer is effected by radiation and conduction. A combination of the decomposition and the heat transfer model allows the calculation of melt front propagation rates dependent upon the bulk temperature of the core melt.

- 4993 MODELING THE MECHANICAL RESPONSE OF A DEGRADED REACTOR FUEL CLAD. Subbaraman, G. (Inst. for Resource Management Inc., Bethesda, MD); Reifsnider, K.L. pp 171-181 of Environmental degradation of engineering materials. Louthan, M.R. Jr.; McNitt, R.P. (eds.). Blacksburg, VA; Virginia Polytechnic Inst. and State Univ. (1977).

From Conference on environmental degradation of engineering materials; Blacksburg, VA, USA (10 Oct 1977).

In-service degradation is known to occur in fuel clads of reactor systems currently in operation and under design. Breach and failure of fuel clads because of degradation can lead to operational difficulties and closing down of reactors. The failure of a few fuel pins of a subassembly may lead to propagation of failures in other subassemblies due to thermal and pressure induced blanketing effects. From a mechanics analysis standpoint it is imperative that the degradation process be quantitatively described so that conservative selection of materials and design bases can be employed to avoid failure. The degradation of the LMFBR fuel clad (stainless steel Type 316) subjected to strong property variations due to the operating liquid sodium environment, and the elastic, elastic-plastic mechanics analysis of the clad is described, including these variations.

- 4994 MONTE CARLO APPROACH TO THE THEORETICAL PREDICTION OF TEMPERATURE NOISE IN LMFBR SUB-ASSEMBLIES. Firth, D. (UKAEA Reactor Group, Risley). pp 527-542 of Reactor noise - SMORN II. Williams, M.M.R. (Queen Mary Coll., London (UK). Dept. of Nuclear Engineering) (ed.). Oxford, England; Pergamon Press Ltd. (1977).

From Specialists meeting on reactor noise; Gatlinburg, TN, USA (19 Sep 1977).

A candidate technique for detecting incipient blockages in the fuel sub-assemblies of liquid-metal-cooled fast-breeder reactors is the measurement of coolant outlet temperature fluctuations caused by turbulence. A theoretical basis for the method is required, and one model is discussed. The model requires the paths of individual fluid particles to be traced, allowing their motions to vary randomly, subject to certain constraints of the turbulent motion. It is a Monte Carlo method, already used successfully in predicting the transport of scalar fluctuations. Its advantage is that it allows actual time-dependent temperatures to be computed, and so various methods of analysis (power spectral density, amplitude probability distribution, for example) can be examined.

- 4995 NEUTRON NOISE INDUCED BY VIBRATION ON THE FRENCH PHENIX LMFBR. Tigeot, T. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Div. d'Etude et de Developpement des Reacteurs); Le Guillou, G. pp 487-496 of Reactor noise - SMORN II. Williams, M.M.R. (Queen Mary Coll., London (UK). Dept. of Nuclear Engineering) (ed.). Oxford, England; Pergamon Press Ltd. (1977).

From Specialists meeting on reactor noise; Gatlinburg, TN, USA (19 Sep 1977).

The vibratory measurements which are available on the reactor have simultaneously been recorded with the fluctuations of neutronic power measurements. The following were observed: 1) a movement of the carrier concrete slab of the primary circuit at the fundamental frequency corresponding to the pumps rotary speed; 2) a low level vibration on a control rod drive corresponding to an incipient defect; 3) vibrations emergence on the core cover during a working cycle; and 4) all the time, a resonance corresponding to the first vibrational mode of the fuel assemblies under the cooling sodium flow. The evidence of the coherence function between the available neutronic noise measurements and the vibratory transducers. From the experience gained on PHENIX vibrations measurements are included in the SUPER PHENIX monitoring system.

- 4996 CROSS SECTIONS REQUIREMENTS FOR THE PREDICTION OF GAS PRODUCTION RATES FOR USE IN IRRADIATION DAMAGE STUDIES IN LMFBR. Ganesan, S. (Reactor Research Centre, Kalpakkam, India). J. Mysore Univ., Sect. B. Sci.; 26: 261-266(1977). (CONF-7606188--).

From National symposium on radiation physics; Mysore, India (10 Jun 1976).

The development of competitive commercial fast power reactors requires the achievement of long burnup of the fuel. To achieve the required burnup of the fuel, the unfavourable effects of radiation damage must be minimized and materials which resist radiation damage need to be developed. The production of gasses by transmutation reactions in the various constituents of stainless steel leads to deviations from normal mechanical properties and thus becomes an important factor in determining the life time of the material in fast power reactors. The helium that is produced by (n, α) reactions in structural elements can serve as nuclei for void formation. Helium production rates are calculated.

- 4997 STUDY OF THE CONTRIBUTION TO DOPPLER EFFECT FROM STRUCTURAL MATERIALS IN FAST POWER REACTORS. Sharma, M.L.; Ganesan, S. (Reactor Research Centre, Kalpakkam, India). J. Mysore Univ., Sect. B. Sci.; 26: 228-231(1977). (CONF-7606188--).

From National symposium on radiation physics; Mysore, India (10 Jun 1976).

The Doppler coefficient of reactivity is the only prompt reactivity coefficient in the considerations of the safety aspects of large fast power reactors during rapid power transients. Recently the delayed negative reactivity feed back due to the Doppler broadening of the resonances belonging to the structural elements like Ni, Mo, Cr and Fe has been found to be of some importance relative to the contribution of fissile material present in the reactor. The contribution of the structural elements to the overall Doppler coefficient of reactivity for two typical fast critical assemblies SEFOR and ZPPR-2 is analyzed. The choice for the SEFOR core in our analysis is because of the fact that SEFOR reactor was

mainly constructed for studying the Doppler effect in fast reactors and ZPPR-2 was chosen as it has a flattened, two enrichment zone cores which simulate the relative composition of PuO_2/UO_2 , steel and Na of demonstration size (350 to 500 MWe) fast reactor.

- 4998 RESULTS OF THE TEMPERATURE NOISE MEASUREMENTS IN SUBCHANNELS OF SODIUM COOLED ELECTRICALLY HEATED 28 ROD BUNDLE BEHIND A BLOCKAGE. Tuerkcan, E. (Stichting Energieonderzoek Centrum Nederland, Petten). pp 497-506 of Reactor noise - SMORN II. Williams, M.M.R. (Queen Mary Coll., London (UK). Dept. of Nuclear Engineering) (ed.). Oxford, England; Pergamon Press Ltd. (1977).

From Specialists meeting on reactor noise; Gatlinburg, TN, USA (19 Sep 1977).

For LMFBR safety studies a 28 rod bundle has been built at Petten (cooperation of GfK and ECN), representing a 60-degrees section of an SNR-300 fuel element having a 70% flat type central blockage. The aims of the temperature noise measurements were to determine the subchannel coolant velocities behind the blockage to study the mixing of coolant in subchannels of different temperature from behind the blockage to the outlet and to study the temperature noise due to boiling in a subchannel. The temperature noise measurements were carried out in parallel to the other measurements (temperature distribution, etc.), using signals of fourteen subchannel thermocouples placed in five measuring planes behind the blockage. The single phase measurements were made with several heat fluxes (5 W/cm^2 to 120 W/cm^2) inlet flows (0.25 to 3 m/s) and inlet temperatures (250°C to 600°C). Two phase flow is initiated and sustained either by a slow continuous pressure reduction or by stepwise reduction of the main flow. The temperature noise signals were amplified and recorded in analog form. Later the signals were digitized and analyzed by digital computers.

- 4999 BABCOCK AND WILCOX PLUTONIUM FUEL FABRICATION EXPERIENCE. Ross, W.J.; Benson, E.M. (Babcock and Wilcox Co., Apollo, PA). pp III.4.1-III.4.11 of Proceedings of the topical meeting on the plutonium fuel cycle. La Grange Park, IL; American Nuclear Society (1977).

From Conference on the plutonium fuel recycle; Miami Beach, FL, USA (2 May 1977).

The Babcock and Wilcox Plutonium Fuels Plant has fabricated 20,721 LMFBR type fuel pins of 25 w/o PuO_2 - UO_2 fuel for Core 1 of FFTF and is currently fabricating 35,500 additional fuel pins for Cores 3 and 4. LWR fuel pellets containing 2.6% PuO_2 - UO_2 were also fabricated, for in-reactor densification tests for EPRI, using equipment and the process considered typical for a 75 ton LWR plutonium fuel fabrication line. B and W's experience with the commercial plutonium fuel fabrication work has shown that a continuing effort must be directed toward an effective waste management program. The generation of waste must be minimized and controlled during the fabrication campaign to reduce the financial impact on the program.

- 5000 STUDY OF THE DISSOLUTION OF REFRACTORY PuO_2 IN AQUEOUS HYDRIODIC ACID SOLUTIONS. Tallent, O.K.; Mailen, J.C.; Williams, K.D. (Oak Ridge National Lab., TN). pp VI.4.1-VI.4.13 of Proceedings of the topical meeting on plutonium fuel cycle. La Grange Park, IL; American Nuclear Society (1977).

From Conference on the plutonium fuel recycle; Miami Beach, FL, USA (2 May 1977).

The dissolution of refractory PuO_2 in aqueous HI solution was studied as a part of a program to develop a suitable dissolution method for small amounts of PuO_2 remaining in residual solids after HNO_3 dissolution of LWR or LMFBR fuels. The PuO_2 dissolution reaction in HI was found to be second order with respect to mean HI activity and first order with respect to PuO_2 surface area. The apparent energy of activation for the dissolution reaction was found to be 13.4 kcal. The dissolved plutonium was found to be essentially all Pu(III) . Dissolution rates were found to remain relatively constant with time compared to the decreasing rates obtained in HNO_3 -HF- H_2O dissolvents.

- 5001 FULL SCALE TESTS AND THERMAL DESIGN CORRELATION FOR COILED ONCE-THROUGH STEAM GENERATORS. Campolunghi, F.; Cumo, M.; Ferrari, G.; Palazzi, G. (C.N.E.N., Rome, Italy). AICHE Symp. Ser.; 73: No. 164, 215-227(1977). (CONF-760816--).

From National heat transfer conference; St. Louis, MO, USA (8 Aug 1976).

Full scale heat transfer tests have been performed in a coiled once-through steam generating tube ($d_{\text{sub tube}} = 0.0155 \text{ m}$, $D_{\text{sub coil}} = 0.836 \text{ m}$), with subcooled water at inlet and superheated steam at outlet. Pressure (p) ranges from 80 to 170 bar, specific mass flowrate (G) from 1000 to 2500 $\text{kg}/\text{m}^2\text{s}$, surface heat flux (q'') from 10 to 30 $\times 10^4 \text{ W}/\text{m}^2$ q''/G ratios have been kept almost constant, close to a typical value for LMFBR's steam generators: q''/G approx. = 0.13 kJ/kg . It has been deduced that for $q''/\text{G} < 0.13 \text{ kJ}/\text{kg}$ no DNB occurs. Thus designers may consider a boiling length extended up to 100% quality. Along this length (an important one of steam generators) a mean boiling heat transfer coefficient is suggested: $h_{\text{sub b}} = 11.226 q''^{0.6} e^{0.0132 - \text{sub p}} / (\text{W}/\text{m}^2\text{°C})$ within a $\pm 30\%$ error margin. Internal and external side heat transfer coefficients, in the superheated steam region of the coil, may differ up to 100% and more.

- 5002 EFFECTS OF NONLINEAR OR 'NOT-LINEAR' CHARACTERISTICS OF REACTOR KINETICS ON POWER REACTOR NOISE ANALYSIS AND PROPOSAL OF APPLICATION OF SOME NEWLY DEVELOPING TECHNIQUES TO POWER REACTOR NOISE ANALYSIS. Kataoka, H. Prog. Nucl. Energy; 1: No. 2-4, 575-582(1977).

Since noise analysis techniques were proposed, they have been useful for subcritical assemblies and zero-power reactors, but not enough successful for power-reactors, because fundamental theory of power-reactor noise is not established yet. Despite various kinds of trial theory for understanding this discrepancy, none of them could get the final solution. Therefore, the author has proposed consideration of the effect of so-called 'not-linear' kinetics or special type of nonlinear kinetics combining the feedback in the nuclear reactor. The author shows many difficulties in the application of conventional noise analysis techniques to power reactors; and in order to overcome these difficulties, proposes the necessity and usefulness of the application of some newly-developing analysis-techniques. These techniques could be effective for the times both before and after high-power-reactor noise theory is established. 25 refs.

- 5003 THERMAL-HYDRAULIC EVALUATION OF ADVANCED WIRE-WRAP ASSEMBLIES. Wei, J.P. (General Electric Co., Sunnyvale, CA). AICHE Symp. Ser.; 73: No. 164, 105-113(1977). (CONF-760816--).

From National heat transfer conference; St.

Louis, MO, USA (8 Aug 1976).

Fuel assemblies for liquid-metal-cooled fast breeder reactors consist of a large number of rods uniformly spaced by wire wrap in a hexagonal array. Thermal-hydraulic analyses of a 217-rod wire-wrapped assembly with different wire-wrap geometries was accomplished. Advanced designs considered are staggered starts, distributed wireless fuel rods and locked wrap.

- 5004 EMPIRICAL METHOD TO CALCULATE CLINCH RIVER BREEDER REACTOR PLANT (CRBRP) INLET PLENUM TRANSIENT TEMPERATURES. Howarth, W.L. (Westinghouse Electric Corp., Madison, PA). AICHE Symp. Ser.; 73: No. 164, 114-119(1977). (CONF-760816--).

From National heat transfer conference; St. Louis, MO, USA (8 Aug 1976).

Sodium flow enters the CRBRP inlet plenum via three inlet loops. An empirical equation was formulated to calculate transient temperatures in the inlet plenum from known inlet loop flows and temperatures. The constants in the empirical equation were measured in a 1/4 scale Inlet Plenum Model using water as the test fluid and an electrolyte to simulate the sodium temperature distribution. Step electrolyte transients at 100% model flow were used to calculate the equation constants. The empirical equation was checked by comparing its results with test results from additional tests that covered the inlet plenum operational range.

- 5005 INFLUENCE OF SCALE SIZE AND FLUID THERMAL PROPERTIES IN SIMULATING LMFBR OUTLET PLENUM BEHAVIOR. Lorenz, J.J.; Carlson, R.D.; Howard, P.A. (Argonne National Lab., IL). AICHE Symp. Ser.; 73: No. 164, 120-126(1977). (CONF-760816--).

From National heat transfer conference; St. Louis, MO, USA (8 Aug 1976).

The influence of scale size and fluid thermal properties in simulating LMFBR outlet plenum behavior was studied. Tests were performed with water and sodium using 1/15-scale models of LMFBR outlet plenums. It was found that differences in the behavior of sodium and water were relatively modest despite a large variance in the respective Peclet numbers. The effect of scale size was found to be unimportant as evidenced by the reasonably good overall agreement of the present 1/15-scale results and 0.55-scale data of Carr and Flanagan. It was concluded that for the purpose of obtaining engineering design data relating to LMFBR outlet plenums, small scale water tests will provide a fairly good direct simulation of full scale sodium behavior.

- 5006 PREDICTIVE TECHNIQUE FOR TEMPERATURE AND EDDY DIFFUSIVITY DISTRIBUTIONS IN LIQUID METAL TURBULENT PIPE FLOW. Klein, G.A.; Sasonke, A. (Purdue Univ., West Lafayette, IN). AICHE Symp. Ser.; 73: No. 164, 127-132(1977). (CONF-760816--).

From National heat transfer conference; St. Louis, MO, USA (8 Aug 1976).

A semi-empirical technique was developed for determining the eddy diffusivity of heat, ϵ /sub H/, independent of temperature gradient measurements for liquid metals. Two functions represent ϵ /sub H/ in the low and high Reynolds number range, respectively. These describe available data and can be refined as additional results become available.

- 5007 MEASUREMENTS OF SUBASSEMBLY AND CORE TEMPERATURE DISTRIBUTIONS IN AN LMFBR. Singer, R.M.; Gillette, J.L. (Argonne

National Lab., IL). AICHE Symp. Ser.; 73: No. 164, 97-104(1977). (CONF-760816--).

From National heat transfer conference; St. Louis, MO, USA (8 Aug 1976).

In the analyses of temperature distributions in the core of a liquid metal cooled fast breeder reactor (LMFBR) under both normal and abnormal operating conditions, it is generally assumed that there is little, if any, thermal interaction between adjacent subassemblies. Since intersubassembly heat transfer tends to reduce the radial temperature gradient in a reactor, thereby ameliorating the effects of local overheating, this assumption is conservative. In order to assess the importance of this effect as well as that of flow redistribution in a reactor core, a series of in-core and subassembly outlet measurements were made in EBR-II over a wide range of operating power and flow conditions, encompassing both forced and natural convection. The results of this study indicate that radial heat transfer and flow redistribution are important mechanisms which should be taken into account in the thermal-hydraulic analyses of LMFBR cores, especially under low flow rate conditions.

- 5008 RADIATION-DAMAGE CONSIDERATIONS. Frost, B.R.T. (Argonne National Lab., IL). pp 521-543 of Fundamental aspects of structural alloy design. Jaffee, R.I.; Wilcox, B.A. (eds.). New York, NY; Plenum Publishing Corp. (1977).

The designs of nuclear-fission and fusion power plants do not, in general, appear to make unusual demands on materials in terms of mechanical-property requirements. The needs of the designer can be met by commercially available alloys. However, the radiation environment produces unique effects on the composition, microstructure, and defect population of these alloys, resulting in time-dependent and time-independent changes in mechanical properties. The material needs of the core of a Liquid-Metal Fast-Breeder Reactor (LMFBR) and of the first wall of a fusion reactor are discussed. In the case of the LMFBR core, the phenomenon of void swelling causes serious design problems, and a search is being made for a low-swelling alloy that has adequate mechanical properties. The fusion reactor poses different problems because the neutron energy is high (14 MeV) and is accompanied by a high flux of charged particles. The long-term choices for a wall material have been narrowed to vanadium and niobium alloys. In the search for low-swelling alloys, it has become clear that minor elements play an important role in determining the nature of the radiation effects. The segregation of minor elements to void surfaces and the dispersion and reformation of second-phase precipitates are two important radiation-induced phenomena that require additional study in view of their influence on void swelling and high-temperature properties.

- 5009 DETECTION OF SODIUM VAPOR BUBBLE COLLAPSE IN A LIQUID METAL FAST BREEDER REACTOR. Carey, W.M.; Gavin, A.P.; Bobis, J.P.; Sheen, S.H.; Anderson, I.T.; Doolittle, R.D.; Albrecht, R.W. (Argonne National Lab., Ill. (USA)). pp 437-468 of Reactor noise - SMORN II. Proceedings of the 2. specialists meeting on reactor noise, Gatlinburg, Tennessee, USA 19-23 September 1977. Williams, M.M.R. (Queen Mary Coll., London (UK). Dept. of Nuclear Engineering) (ed.). Oxford; Pergamon Press Ltd. (1977).

Sodium boiling detection utilizing the sound pressure emanated during the collapse of a

sodium vapour bubble in a subcooled media is discussed in terms of the sound characteristic, the reactor ambient noise background, transmission loss considerations and performance criteria. Data obtained in several loss of flow experiments on Fast Test Reactor Fuel Elements indicate that the collapse of the sodium vapour bubble depends on the presence of a subcooled structure or sodium. The collapse pressure pulse was observed in all cases to be on the order of a kPa, indicating a soft type of cavitation collapse. Spectral examination of the pulses indicates the response function of the test structure and geometry is important. The sodium boiling observed in these experiments was observed to occur at a low ($<50^{\circ}\text{C}$) liquid superheat with the rate of occurrence of sodium vapor bubble collapse in the 3 to 30 Hz range. Reactor ambient noise data were found to be due to machinery induced vibrations flow induced vibrations, and flow noise. These data were further found to be weakly stationary enhancing the possibility of acoustic surveillance of an operating Liquid Metal Fast Breeder Reactor. Based on these noise characteristics and extrapolating the noise measurements from the Fast Flux Test Facility Pump (FFTP), one would expect a signal to noise ratio of up to 20 dB in the absence of transmission loss. The requirement of a low false alarm probability is shown to necessitate post detection analysis of the collapse event sequence and the cross correlation with the second derivative of the neutronic boiling detection signal. Sodium boiling detection using the sounds emitted during sodium vapor bubble collapse are shown to be feasible but a need for in-reactor demonstration is necessary.

- 5010 DETERMINATION OF THERMOCOUPLE TRANSFER-FUNCTIONS AND FLUID-FLOW VELOCITIES BY TEMPERATURE-NOISE MEASUREMENTS IN LIQUID SODIUM. Benkert, J.; Mika, C.; Raes, K.H.; Stegemann, D. (Technische Univ. Hannover (Germany, F.R.)). pp 553-563 of Reactor noise - SMORN II. Williams, M.M.R. (Queen Mary Coll., London (UK). Dept. of Nuclear Engineering) (ed.). Oxford, England; Pergamon Press Ltd. (1977).

From Specialists meeting on reactor noise; Gatlinburg, TN, USA (19 Sep 1977).

The time response of TI-, TM- and intrinsic thermocouples has been investigated in sodium by determination of frequency dependent thermocouple transfer functions and related delay times. Experimentally temperature fluctuations (temperature noise), generated in sodium by heated channels or injection of hot sodium, have been used as driving sources. The measured transfer functions and delay times have been compared to predictions from theoretical models. Good agreement was found. Intrinsic and TM-type thermocouples are best suited if fast response is required. Delay times in the range of one millisecond or upper break frequencies of 1000 Hz were determined for intrinsic thermocouples. The determination of sodium flow velocities by the transit time correlation technique showed systematic error sources for analysis at low frequencies due to intermediate heat storage between the thermocouple positions. Statistical errors have been determined as function of thermocouple distance, frequency range and time analysis. Minimum errors are obtained for thermocouple distances from 100 to 150 mm and in the frequency range 90 to 120 Hz. A transit time correlator, which automatically tracks the maximum of the cross-correlation function and displays the velocity, has been successfully used for signals with correlation coefficients greater than 0.3.

- 5011 NOISE AND DC BALANCED OUTLET TEMPERATURE SIGNALS FOR MONITORING COOLANT FLOW IN LMFBF FUEL ELEMENTS. Edelmann, M. (Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). pp 543-552 of Reactor noise - SMORN II. Williams, M.M.R. (Queen Mary Coll., London (UK). Dept. of Nuclear Engineering) (ed.). Oxford, England; Pergamon Press Ltd. (1977).

From Specialists meeting on reactor noise; Gatlinburg, TN, USA (19 Sep 1977).

Local cooling disturbances in LMFBF fuel elements may have serious safety implications for the whole reactor core. They have to be detected reliably in an early stage of their formation therefore. This can be accomplished in principle by individual monitoring of the coolant flow rate or the coolant outlet temperature of the sub-assemblies with high precision. In this paper a method is proposed to increase the sensitivity of outlet temperature signals to cooling disturbances. Using balanced temperature signals provides a means for eliminating the normal variations from the original signals which limit the sensitivity and speed of response to cooling disturbances. It is shown that a balanced signal can be derived easily from the original temperature signal by subtracting an inlet temperature and a neutron detector signal with appropriate time shift. The method was tested with tape-recorded noise signals of the KNK I reactor at Karlsruhe. The experimental results confirm the theoretical predictions. A significant reduction of the uncertainty of measured outlet temperatures was achieved. This enables very sensitive and fast response monitoring of coolant flow. Furthermore, it was found that minimizing the variance of the balanced signal offers the possibility for a rough determination of the heat transfer coefficient of the fuel rods during normal reactor operation at power.

- 5012 DETECTION OF LOCAL BOILING IN AN LMFBF SUBASSEMBLY BY TEMPERATURE FLUCTUATIONS ANALYSIS AT THE OUTLET. Krebs, L.; Weinkoetz, G. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorbauelemente). pp 507-515 of Reactor noise - SMORN II. Williams, M.M.R. (Queen Mary Coll., London (UK). Dept. of Nuclear Engineering) (ed.). Oxford, England; Pergamon Press Ltd. (1977).

From Specialists meeting on reactor noise; Gatlinburg, TN, USA (19 Sep 1977).

The aim of the experiments is to detect boiling in a sodium cooled subassembly by measuring fluctuations behind the bundle outlet. The measurements were carried out on an electrically heated 28-rod bundle with a partially blocked section. Fast responding thermocouples were installed downstream of the bundle outlet and downstream of a flow mixing system. Statistical parameters were investigated such as root mean square (RMS) and power spectrum density (PSD). The boiling conditions were generated by reducing the system pressure or flow velocity reduction. The experiments have shown that statistical analysis of temperature fluctuations can produce significant results in the detection of boiling behaviour at both the outlet of a subassembly, and behind a flow mixing system.

- 5013 BOILING DETECTION IN FAST REACTORS BY NOISE ANALYSIS STUDIES PERFORMED IN FRANCE. Le Guillou, G.; Berger, R.; Brunet, M. (CEA Centre d'Etudes Nucleaires de Cadarache, 13 - Saint-Paul-les-Durance (France). Dept. des Reacteurs a Neutrons Rapides). pp 409-426 of

Reactor noise - SMORN II. Williams, M.M.R. (Queen Mary Coll., London (UK). Dept. of Nuclear Engineering) (ed.). Oxford, England; Pergamon Press Ltd. (1977).

From Specialists meeting on reactor noise; Gatlinburg, TN, USA (19 Sep 1977).

The onset of accidental conditions particularly the boiling of sodium is generally related to background noise modifications in an operating reactor. The detection of in-core boiling may be achieved by three different analysis methods: neutron noise, acoustic noise and temperature fluctuations at the subassembly outlets. This paper deals with tests and measurements made in this field. The subject is discussed under the following headings: boiling in the loop (general description of experiments, detection of boiling by monitoring flow and pressure noises, detection of boiling by thermal noise analysis, acoustic detection of boiling); detection of boiling in the core: description of the 'NABO' experiment carried out on the reactor RAPSODIE (brief description of experiment, detection of boiling by neutron noise analysis, detection of boiling by acoustic noise analysis).

5014 ACOUSTIC DETECTION OF BOILING IN LMFBRs: AN ESTIMATE OF SENSITIVITY DERIVED FROM EXPERIMENTS DURING THE COMMISSIONING OF PFR. Macleod, I.D.; Catling, E.; Taylor, C.G. (UKAEA Reactor Group, Springfield). pp 469-485 of Reactor noise - SMORN II. Williams, M.M.R. (Queen Mary Coll., London (UK). Dept. of Nuclear Engineering) (ed.). Oxford, England; Pergamon Press Ltd. (1977).

From Specialists meeting on reactor noise; Gatlinburg, TN, USA (19 Sep 1977).

This paper describes a programme of experiments undertaken during the commissioning of the 600-MW PFR at Dounreay. The objective of the programme was to obtain data on reactor background acoustic noise levels and to estimate the sensitivity of the installed acoustic boiling detection system. The boiling-noise detection system installed in PFR consists of seven waveguides. The waveguides are solid metal rods which extend from the reactor top to just above the core top level. Five are arranged on a 0.75m radius circle with two at the edge of the core. The sensing elements are standard accelerometers. For the commissioning experiments three rigs were installed in the core locations below three of the waveguides. Each rig carried three sodium-proof microphones at levels corresponding to core top, mid-core and core bottom. Each rig also carried an electrically-powered vapour generator which could inject a stream of sodium vapour into the surrounding 250°C sodium, so producing a boiling type noise. In addition to these nine microphones in the core, microphones were also installed at the inlet and outlet of each of the three pumps. To handle the data for this experiment a system of amplifiers and three fourteen-track tape recorders were used. To check amplifier gains a calibration signal was automatically injected at the end of each record. On-line one-third octave and narrow band analysis was also available, but the bulk of the analysis was done from the tape records. The commissioning programme was in two parts: a measurement of background noise and an investigation of the detection sensitivity at the waveguides of the noise from the vapour generator (this implicitly included the transmission loss).

5015 DESIGN RULES FOR HIGH TEMPERATURE PLANT - THE IMPLICATIONS OF RECENT RESEARCH IN RELATION TO CURRENT PRACTICE. Townley, C.H.A. (Central Electricity Generating Board, Berkeley (UK).

Berkeley Nuclear Labs.). pp 1-3 of Recent developments in high temperature design methods. London; Mechanical Engineering Publications for IMechE (1977).

From Conference on recent developments in high temperature design methods; London, UK (6 Dec 1977).

An historical summary is presented of design rules for high temperature plant, leading to the rules applicable to high temperature reactors, particularly the liquid metal fast breeder reactor. Special attention is given to creep rupture properties of ferritic and austenitic materials used for the construction of components such as boilers and pressure vessels.

5016 MATERIALS DATA FOR SEVERE VARIABLE LOADING. Wood, D.S. (UKAEA, Risley. Nuclear Power Development Labs.). pp 65-70 of Recent developments in high temperature design methods. London; Mechanical Engineering Publications for IMechE (1977).

From Conference on recent developments in high temperature design methods; London, UK (6 Dec 1977).

The reversed strain behaviour of structural materials at temperatures in the creep range is an important materials parameter in sodium cooled fast reactors. Existing rules for designing against combined creep/fatigue are suspect due to lack of long term data. A UK programme of creep/fatigue tests is outlined which includes long term tests and a study of cast variations. It is believed that information from this programme should substantially improve our ability to provide design guidance against creep/fatigue damage.

5017 DETECTION OF ACOUSTIC EMISSION EVENTS IN THE PRESENCE OF PARASITIC NOISE. Fuge, R. (Zentralinstitut fuer Kernforschung, Rossendorf bei Dresden (German Democratic Republic)). pp 681-690 of Reactor noise - SMORN II. Williams, M.M.R. (Queen Mary Coll., London (UK). Dept. of Nuclear Engineering) (ed.). Oxford, England; Pergamon Press Ltd. (1977).

From Specialists meeting on reactor noise; Gatlinburg, TN, USA (19 Sep 1977).

The optimum threshold for the detection of acoustic emission bursts, which are masked by parasitic noise is calculated. The theoretical results are applied to the acoustic noise generated by boiling sodium.

5018 MICROSTRUCTURAL ASPECTS OF NUCLEAR FUEL. Graham, R.A.; Rasmussen, D.E.; Carlson, M.C.J. pp 404-413 of Ceramic microstructures '76: with emphasis on energy related applications. Fulrath, R.M.; Pask, J.A. (eds.). Boulder, CO; Westview Press (1977).

From 6. international symposium - ceramic microstructures '76; Berkeley, CA, USA (24 Aug 1976).

Pore size distribution and morphology characterization data were obtained for a wide range of mixed oxide fuel microstructures and related to fabrication parameters. Fuel microhomogeneity is characterized as equivalent sized PuO_2 regions calculated from concentration dependent size distribution information. Techniques described can provide an effective basis for comparing and monitoring effectiveness of alternate fabrication techniques and definition of critical parameters.

5019 APPLICATION OF PROBABILISTIC METHODS TO SAFETY R AND D AND DESIGN CHOICES. Gavigan, F.X.; Griffith, J.D. (Energy Research and

Development Administration, Washington, DC). pp 22-41 of Nuclear systems reliability engineering and risk assessment. Fussell, J.B.; Burdick, G.R. (eds.). Philadelphia, PA; Society for Industrial and Applied Mathematics (1977).

From International conference on nuclear systems reliability engineering and risk assessment; Gatlinburg, TN, USA (20 Jun 1977).

The Liquid Metal Fast Breeder Reactor (LMFBR) safety program is committed to identifying and exploiting areas in which probabilistic methods can be developed and used in making reactor safety R and D choices and optimizing designs of safety systems. Emphasis will be placed on a positive approach of solidifying and expanding our knowledge. This will provide the groundwork for a consensus on FBR risk. The management structure which will be used is based on a mechanistic approach to an LMFBR Core Disruptive Accident (CDA) with risk partitioned into "Lines of Assurance," i.e., independent, phenomenologically-based barriers which will impede or mitigate the progression and consequences of accident sequences. Quantitative determination of the probability of breach of these barriers through the completion of work identified for each Line of Assurance will allow the quantification of the contribution to risk reduction associated with the success of each barrier. This process can lead to better use of resources by channeling R and D in directions which promise the greatest potential for reducing risk and by identifying an orderly approach to the development and demonstration of design features which will keep LMFBR risks at an acceptable level.

5020 PROBABILISTIC APPROACHES TO ADVANCED REACTOR DESIGN OPTIMIZATION. Burdick, G.R. (EG and G Idaho, Inc., Idaho Falls, ID); Rasmuson, D.M.; Weisman, J. pp 395-417 of Nuclear systems reliability engineering and risk assessment. Fussell, J.B.; Burdick, G.R. (eds.). Philadelphia, PA; Society for Industrial and Applied Mathematics (1977).

From International conference on nuclear systems reliability engineering and risk assessment; Gatlinburg, TN, USA (20 Jun 1977).

Approaches to advanced reactor design which combine probabilistic methods with system optimization techniques are proposed. In recognition that the best available design and manufacturing methods have been applied to component design and manufacture, it is suggested that significant improvement of nuclear systems performance is possible only through modification of the systems configuration. Thus, the models used in the optimization procedures are those for redundancy and diversity allocation or models which provide goals for such allocation. It is shown that diversity and redundancy allocation may be improved through the construction and use of functions which represent an estimate of the best cost obtainable for a given level of system availability. The use of functional relationships between cost and availability is extended to the case where acceptable levels of risk have been defined for plant operation. In this case, the cost functions for the various safety systems are used to form an overall objective function. Societal risk expressions are formulated in terms of safety system unavailabilities. Construction of the cost objective function and risk expressions constitutes completion of the top-level problem. A second-level problem is then solved which performs trade-offs of cost and availability among the safety systems. The second-level problem solution is a set of availability goals and cost guidelines which

can be used by designers in solution of third-level problems which are determinations of the actual safety system configurations.

5021 PROCEDURE FOR THE USE OF PROBABILISTIC RISK ASSESSMENTS TO ESTABLISH RANKING OF RESEARCH AND DEVELOPMENT GOALS. Temme, M.I.; El-Sheikh, K.A. (General Electric Co., Sunnyvale, CA). pp 418-442 of Nuclear systems reliability engineering and risk assessment. Fussell, J.B.; Burdick, G.R. (eds.). Philadelphia, PA; Society for Industrial and Applied Mathematics (1977).

From International conference on nuclear systems reliability engineering and risk assessment; Gatlinburg, TN, USA (20 Jun 1977).

A procedure for Ranking Research and Development goals on the basis of their importance to risk is described. Using information developed in a typical LMFBR risk assessment, rules for selection of the most important events or groups of events (e.g., event sequences) are defined. The procedure is illustrated through its application to an example, using a First Cycle Risk Assessment of FFTF.

5022 ROLE OF RELIABILITY AND RISK ASSESSMENT IN LMFBR DESIGN: IMPLEMENTATION OF RELIABILITY IN LMFBR DESIGN. Graham, J.; Zemanick, P.P. (Westinghouse Electric Co., Madison, PA). pp 686-709 of Nuclear systems reliability engineering and risk assessment. Fussell, J.B.; Burdick, G.R. (eds.). Philadelphia, PA; Society for Industrial and Applied Mathematics (1977).

From International conference on nuclear systems reliability engineering and risk assessment; Gatlinburg, TN, USA (20 Jun 1977).

This paper better concerns future developments in LMFBR licensing technology. Federal Regulations (10 CFR 50.34) require that the preliminary safety analysis provide analysis and evaluation "with the objective of assessing the risk to public health and safety" to determine margins of safety and the adequacy of the plant. Hitherto, the assessment of risk has been qualitative but it has become increasingly apparent that quantitative assessments would provide a better basis for judgement. Potential future roles of reliability and risk assessment are discussed in the context of providing additional confirmation of the safety of LMFBR designs. Potential acceptability criteria for risk evaluations are outlined. The reliability implications of designing components to the ASME Code Section III requirements are discussed. General judgements are provided as well as the preliminary results of probabilistic studies of selected specific limits. There is a different reliability significance for the mandatory rules for normal, upset, and emergency conditions versus the non-mandatory rules for normal, upset, and emergency conditions versus the non-mandatory guidance for faulted conditions.

5023 ON THE DEVELOPMENT OF NUCLEAR POWER. Emel'yanov, I.Ya. Power Eng.; 15: No. 5, 10-26(1977).

A brief exposition is given of experience in the development of nuclear power in the USSR beginning with the first nuclear reactor and the first atomic power station (APS) in the world and extending to large-scale construction of commercial APS. New trends and problems are characterized which have developed in nuclear power engineering at the contemporary stage of its development. Certain problems in the functioning of APS in combined electric power

systems are considered along with general problems of the utilization of nuclear reactors for centralized industrial and household heat supply, as well as for the realization of high-temperature industrial technology processes in the chemical industry, ferrous metallurgy, in the conversion of energy carriers, and in the production of hydrogen. Certain economic-ecological aspects of the development of nuclear power are elucidated in perspective.

- 5024 APPLICATION OF MAGNETOHYDRODYNAMIC GENERATORS IN NUCLEAR POWER ENGINEERING. Kirillin, A.; Sheindlin, A.E. Power Eng.; 15: No. 5, 27-34(1977).

The prospects for applying magnetohydrodynamic generators in nuclear power engineering in combination with various types of nuclear reactors are considered. The possibilities of combining liquid-metal fast-neutron reactors with liquid-metal MHD generators, high-temperature fission and fusion reactors with MHD generators operating on nonequilibrium plasma, and gas-phase fission reactors with MHD generators operating on uranium hexafluoride plasma are evaluated. It is concluded that MHD generators operating on a closed cycle may find fairly extensive application in nuclear power engineering.

- 5025 CRACK INITIATION AND PROPAGATION AS A CONSEQUENCE OF THERMAL SHOCK. Lloyd, G.J.; Wood, D.S. (UKAEA, Risley. Nuclear Power Development Establishment). pp vp of Structural mechanics in reactor technology. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
CRACKS;ELASTICITY;FATIGUE;FRACTURES;LMFBR TYPE REACTORS;MECHANICS;STAINLESS STEEL-316; STRAINS;STRESS INTENSITY FACTORS;STRESSES; THERMAL SHOCK

- 5026 FLOW-INDUCED VIBRATION SCALE MODEL TEST OF THE CRBR INSTRUMENT POST. Mulcahy, T.M.; Yeh, T.T.; Lawrence, W.P. (Argonne National Lab., IL). pp 59-66 of Fluid structure interaction phenomena in pressure vessel and piping systems. Au-Yang, M.K.; Brown, S.J. Jr. (eds.). New York, NY; American Society of Mechanical Engineers (1977).
From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).
The analysis is presented for the Clinch River Breeder Reactor Instrument Post which has a finned rocket-like geometry with high potential for excitation. Correlated shedding of vortices from the trailing edge of plates in smooth, parallel flow is well documented. The basis of the excitation mechanism is discussed as well as scaling criteria and distortions, model construction and instrumentation, and selected results. No indication of correlated vortex shedding was observed in the response.

- 5027 NUMERICAL FLUID-STRUCTURE INTERACTION ANALYSIS OF PIPING SYSTEMS. A-Moneim, M.T. (Argonne National Lab., IL). pp 107-121 of Fluid structure interaction phenomena in pressure vessel and piping systems. Au-Yang, M.K.; Brown, S.J. Jr. (eds.). New York, NY; American Society of Mechanical Engineers (1977).
From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

A computer code for transient two-

dimensional coupled hydrodynamic-structural response analysis of piping systems under the effect of internally travelling pressure pulses is described. Two dimensional hydrodynamic models for straight pipes, elbows, generalized piping component, tee branching junctions, and surge tanks are included and coupled together to form a general piping system that can be subjected to simultaneous pressure pulses of the form $P(t)$. Options are provided in the generalized piping component; namely, interior rigid wall simulation, nine different exterior wall configurations, and heat exchanger tube bundle effects. Thus, it can be used in modelling of valves, pipe flow area changes and heat exchangers.

- 5028 PRESSURE TRANSIENT ANALYSIS IN SINGLE AND TWO-PHASE WATER BY FINITE DIFFERENCE METHODS. Berry, G.F.; Daley, J.G. (Argonne National Lab., IL (USA)). pp 1-12 of Structural mechanics in reactor technology. Vol. B. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
An important consideration in the design of LMFBR steam generators is the possibility of leakage from a steam generator water tube. The ensuing sodium/water reaction will be largely controlled by the amount of water available at the leak site, thus analysis methods treating this event must have the capability of accurately modeling pressure transients through all states of water occurring in a steam generator, whether single or two-phase. The equation systems of the present model consist of the conservation equations together with an equation of state for one-dimensional homogeneous flow. These equations are then solved using finite difference technique with phase considerations and non-equilibrium effects being treated through the equation of state. Fluid-structure interactions were not included at the time of this writing but are treated in the systems code which will contain this water-side transient model.

- 5029 INVESTIGATION OF THE STRUCTURAL INTEGRITY OF THE REACTOR VESSEL FOR A POSTULATED CORE MELTDOWN ACCIDENT. Glueckler, E.L. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.); Clever, R.M.; Rumble, E.T. (Science Applications, Inc., Palo Alto, CA (USA)). pp 1-12 of Structural mechanics in reactor technology. Vol. G. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
If a core meltdown accident is postulated to occur, a large mass of molten fuel and steel debris could collect at the bottom of the reactor vessel. As a consequence, the vessel would be subjected to rapid heating and surface melting. In the absence of adequate cooling, inelastic failure or meltthrough could occur. The two dimensional thermal response of the vessel was obtained from a multiregion ADI solution of the fully implicit finite difference analogues of the governing differential equations. These equations have been transformed into a coordinate system moving with the melting surface. All non-linear effects such as the surface boundary conditions were included using an iterative scheme with quasi-linear equations.

- 5030 DAMPING OF THE RADIAL IMPULSIVE MOTION

OF LMFBR CORE COMPONENTS SEPARATED BY FLUID SQUEEZE FILMS. Liebe, R.; Zehlein, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). pp 1-12 of Structural mechanics in reactor technology. Vol. B. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The core deformation of a liquid metal cooled fast breeder reactor (LMFBR) due to local pressure propagation from rapid energy releases is a complex three-dimensional fluid-structure-interaction problem. High pressure transients of short duration cause structural deformation of the closely spaced fuel elements, which are surrounded by the flowing coolant. Corresponding relative displacements give rise to squeezing fluid motion in the thin layers between the subassemblies. Therefore significant backpressures are produced and the resulting time and space dependent fluid forces are acting on the structure as additional non-conservative external loads. Realistic LMFBR safety analysis of several clustered fuel elements have to account for such flow induced forces. Several idealized models have been proposed to study some aspects of the complex problem. As part of the core mechanics activities at GfK Karlsruhe this paper describes two fluid flow models (model A, model B), which are shown to be suitable for physically coupled fluid-structure analyses. Important assumptions are discussed in both cases and basic equations are derived for one- and two-dimensional incompressible flow fields. The interface of corresponding computer codes FLUF (model A) and FLOWAX (model B) with structural dynamics programs is outlined. Finally fluid-structure interaction problems relevant to LMFBR design are analyzed; parametric studies indicate a significant cushioning effect, energy dissipation and a strongly nonlinear as well as time dependent damping of the structural response.

- 5031 REVIEW OF THE ACOUSTIC DETECTION OF BOILING IN NUCLEAR REACTORS. Scarton, H.A. (Rensselaer Polytechnic Inst., Troy, NY); McDonald, J.F.; Carey, W.M. pp 109-123 of Noise and fluids engineering. Hickling, R. (ed.). New York, NY: American Society of Mechanical Engineers (1977).

From Winter annual meeting of the American Society of Mechanical Engineers; Atlanta, GA, USA (27 Nov 1977).

A review is presented of over 260 references on the physics of boiling, stressing acoustic monitoring techniques. Recent experimental results are examined for sodium boiling during LMFBR excursions and the characteristics of the physical acoustic signatures are identified.

- 5032 EULERIAN ALGORITHM FOR ANALYZING FLUID-STRUCTURE INTERACTION IN THE FAST REACTOR CONTAINMENT. Wang, C.Y.; Chang, Y.W.; Fistedis, S.H. (Argonne National Lab., IL (USA)). pp E1/3 1-11 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper presents an algorithm for analyzing the coupled fluid-structure problems in the LMFBR consisting of various internals and primary vessel. The aim is at the development of a generalized Eulerian hydrodynamic scheme, which can be used to link

together the ICE technique with the finite-element structural dynamics method to provide sliding capability at interface, modeling of perforated structure, and analyses of fluid motions at the geometry discontinuity. Since the finite-element method has been already well developed, the work described concentrates on the fluid-structure interaction. The algorithm uses the Eulerian description in the hydrodynamic formulation principally because it has the deciding advantage of treating excessive distortion. It employs the implicit temporal integration scheme to solve the nonlinear, time-dependent equations of fluid mechanics. This scheme not only insures computational stability, but also eases the Courant condition on the time step. Furthermore, the implicit treatment of pressure and velocity fields allows the presence of all flow-speed regimes. Hence, the algorithm is applicable to any degree of excursion. Two sample problems are given, concerning wave propagation in a typical LMFBR which consists of radial shield, core barrel, core-support structure, and the primary vessel.

- 5033 FRACTURE OF LONGITUDINALLY CRACKED DUCTILE TUBES. Bernard, J.; Larsson, H.; Reynen, J. (Commission of the European Communities, Ispra (Italy). Joint Research Centre). pp F7/3 of Structural mechanics in reactor technology. Vol. F. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

CRACKS;DUCTILITY;FINITE ELEMENT METHOD; FRACTURE PROPERTIES;LMFBR TYPE REACTORS; STAINLESS STEEL-304;TUBES

- 5034 VAPOR TRANSPORT AND PRESSURE RISE BEHIND THE LINER OF REACTOR CONTAINMENTS. Kordina, K.; Schneider, U. (Technische Univ. Braunschweig (Germany, F.R.)). pp E6/3 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

CONTAINMENT SYSTEMS;EVAPORATION;HIGH TEMPERATURE;LINERS;LMFBR TYPE REACTORS; PRESTRESSED CONCRETE;REACTOR VESSELS;REINFORCED CONCRETE;STEELS;VAPORS;PRESSURE GRADIENTS

- 5035 RESPONSE OF WATER-FILLED PIPES TO PRESSURE PULSES. Romander, C.M.; Cagliostro, D.J.; Florence, A.L.; Gran, J.K. (Stanford Research Inst., Menlo Park, CA (USA). Poulter Labs.). pp E4/7 1-11 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Argonne National Laboratory has developed two-dimensional axisymmetric fluid-structure-interaction codes to predict the pressures and strains in the primary piping loop caused by HCDA-generated pressure pulses. Verification of these codes requires simple, well-defined experiments in which scale models of the primary piping are loaded with pressure pulses and the pressures, strains, and deformation in the pipe are measured. This paper presents the results of such an experimental program. Experiments were performed on 5-foot-long (3 inch OD, 0.065 inch wall) Ni 200 pipe sections filled with water. Experimental results are

compared with predictions from a one-dimensional, method-of-characteristics computer analysis of pressure pulses propagating in a plastically deforming pipe.

- 5036 DDT - A 3-DIMENSIONAL PROGRAM FOR THE ANALYSIS OF BOWED REACTOR CORES. Menssen, R. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp D2/4 1-12 of Structural mechanics in reactor technology. Vol. D. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The influence of temperature gradients, neutron fluence gradients and irradiation induced creep leads to individually different bowing effects of all elements in a core of a fast reactor. Subsequent contacts between neighbouring elements result in forces of different value. Eventually all elements will arrange in such a way, that an equilibrium of forces between all elements is achieved. The determination of this equilibrium configuration in different operational states of the reactor is of high significance especially for safety aspects. For this reason, in addition to existing 2-dimensional codes, a 3-dimensional code was developed to allow a good description of the whole core bowing behaviour. Because of nonlinearities, standard mathematical techniques for solving linear structural problems are not directly applicable. So an iterative procedure is applied to achieve the equilibrium force configuration. Each iteration step corrects the largest force discrepancy between neighbouring elements. This technique requires a long computer running time to converge, but the locations of gaps and contacts normally do not change if a rough equilibrium is found.

- 5037 CHARACTERIZATION OF RESIDUAL STRESS PROFILES IN IRRADIATED FUEL PIN CLADDING. Nygren, R.E. (Hanford Engineering Development Lab., Richland, WA (USA)). pp C2/5 1-9 of Structural mechanics in reactor technology. Vol. C. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
- From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

During reactor operation a radial temperature gradient exists through the fuel pin cladding. The temperature gradient causes stress through differential thermal expansion and differential swelling. This paper describes a method for determining actual residual stress profiles in cladding. Work of this nature is currently in progress. The experimental method begins by measuring the diameter of a tubular specimen and then slitting the wall of the specimen axially and remeasuring the diameter. Successive, uniform layers of material are removed from the tube wall and after each increment of wall thinning the diameter is measured. The experimental goal is to provide enough data to describe the free deflection ΔD as a function of the wall thinning. Two specific applications for characterizing the residual stress in irradiated fuel pin cladding with limited, discrete data are developed in the paper. One represents a minimum effort approach and requires only four pairs of data. The end product is a 'curve' consisting of four linear segments that approximates the residual stress profile of the cladding specimen. The second technique is a semi-analytical technique for statistically fitting an assumed curve form to the data.

- 5038 DYNAMIC PROPERTIES OF FUEL PINS. Konstantinidis, S. Acta Polytech. (Prague), IV; No. 3, 97-111(1977). (In Czech).

The dynamic properties of fuel pins exposed to vibration by a liquid sodium flow were studied on the model of a fuel element designed for a sodium cooled fast reactor. The test pins consisted of thin-walled steel tubes 7 mm in diameter, 2310 mm in length, and 0.5 mm in wall thickness filled in two thirds of their height with cylindrical pellets 5 mm in diameter and 10 mm in height, or with another material. A model element comprised 37 pins in a hexagonal lattice, spacing was provided using a wire spiral 1 mm in diameter and 100+5 mm in pitch. A mathematical model is described for studying vibration from the point of view of the exciting force analysis and the system vibration. The natural frequency surface tension amplitudes and vibration amplitudes were measured tensometrically. Piezoelectric contact sensors were used for measuring natural frequencies and resistance-type sensors were used for measuring vibration attenuation. Some results are presented of measuring and calculating the pin natural frequencies in variation with internal positive pressure and the relative vibration attenuation of the pins in variation with a different pin fill.

- 5039 VALIDATION OF EXPLOSION CONTAINMENT CODES: ANALYSIS OF BARE RIGID MODELS IN THE COVA PROGRAMME. Benuzzi, A.; Delaval, M.; Jones, A.; Yerkess, A. (Commission of the European Communities, Ispra (Italy). Joint Research Centre); Persson, B. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)); Cameron, I.G.; Hoskin, N.E. (Ministry of Defence, Aldermaston (UK). Atomic Weapons Research Establishment); Lancefield, M.J. (UKAEA, Winfrith. Atomic Energy Establishment). pp E2/3 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

COMPUTER CODES;CONTAINMENT;EXPLOSIONS;FINITE DIFFERENCE METHOD;HYDRODYNAMICS;LMFBR TYPE REACTORS;REACTOR ACCIDENTS;TIME DEPENDENCE

- 5040 DYNAMIC LOADINGS OF SODIUM-WATER REACTIONS IN LMFBR AND FUSION POWER DESIGNS. Chan, C.K. (California Univ., Los Angeles (USA). Dept. of Energy and Kinetics). pp E5/2 1-9 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In liquid metal fast breeder reactor and lithium cooled fusion reactor, a sodium loop is being proposed to transfer heat from the primary coolant loop to the steam turbine cycle. Although by careful design and quality assurance programs, the probability for steam generator tube failure can be minimized, failure will still occur. The direct contact of sodium and water could cause a chemical reaction where hydrogen and sodium compounds are produced. This paper presents an evaluation of the potential hazards as a result of such a reaction. The sodium-water reaction is explosive in nature with highly corrosive chemical by-products. An analytical method is developed to investigate the extent of the reaction zone and the propagation of the

pressure wave in the sodium system.

REACTORS;MOLTEN METAL-WATER REACTIONS;SAFETY;
SODIUM;STEAM GENERATORS

- 5041 STRUCTURAL DYNAMICS IN LMFBR CONTAINMENT ANALYSIS - A BRIEF SURVEY OF COMPUTATIONAL METHODS AND CODES. Chang, Y.W. (Argonne National Lab., IL (USA)). pp E1/1 1-15 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In recent years, the use of computer codes to study the response of primary containment of large, liquid-metal fast breeder reactors (LMFBR) under postulated accident conditions has been adopted by most fast reactor projects. This paper gives a brief survey of the computational methods and codes available for LMFBR containment analysis. The various numerical methods commonly used in the computer codes are compared. It provides the reactor engineers up-to-date information on the development of structural dynamics in LMFBR containment analysis. It can also be used as a basis for the selection of the numerical method in the future code development.

- 5042 MODELING OF REACTOR INTERNALS IN THE HCDA ANALYSIS. Chang, Y.W.; Gvildys, J. (Argonne National Lab., IL (USA)). pp E2/2 1-13 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper deals with the modeling of the reactor internals in the HCDA analysis. Internals which must be included in the analysis are core blankets, shields, fission-gas plenum, core barrel and core support structure. This is because they are placed so close to the reactor core that the expansion of the reactor core is strongly influenced by the presence of these internals. The general approach to the mathematical model is to incorporate, to the extent possible, the actual geometries and the correct material properties. In the HCDA analysis, the mechanism for producing mechanical damage to the primary containment is the expansion of the reactor core. Therefore, it is very important to model the core gas properly. The inertia of the core gas must be included. Four to six zones are needed for a proper representation of the core. The core blankets and shields can also be represented by zones while core barrels must be modeled as thin shells. The core support structure can be modeled either as a lattice of shell elements similar to the actual structure or as an elastic-plastic solid continuum material with overall bending stiffness equivalent to the actual structure. However, the connection between the core support and the reactor vessel must be modeled properly. Finally, to demonstrate the validity of the mathematical model, the results of comparisons with experimental data are presented.

- 5043 FAIL-SAFETY OF LMFBR STEAM GENERATOR SYSTEMS. Chopra, P.S. (Argonne National Lab., IL (USA)). pp E5/3 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

EBR-2 REACTOR;FAILURES;LEAKS;LMFBR TYPE

- 5044 ANALYSIS OF LMFBR CONTAINMENT RESPONSE TO AN HCDA USING A MULTIFIELD EULERIAN CODE. Chu, H.Y.; Chang, Y.W. (Argonne National Lab., IL (USA)). pp E1/2 1-10 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper describes a computer code, MICE (Multifield Implicit Continuous-fluid Eulerian Containment Code), which is being developed at Argonne National Laboratory (ANL) for the analysis of containment response to a hypothetical core disruptive accident (HCDA). The code is applicable to multifield flow problems where material fields are allowed to have penetrations. Reactor structures are treated as axisymmetrical shells and solved by the large-displacement and small-strain theory. Two sample problems have been performed using the MICE code. The first illustrates the relative motions of the material fields after the initiation of a core disassembly accident. Core support structure and core barrel are modeled as rigid obstacles. The second demonstrates the interactions between fluid and structures. Core expansion and reactor wall deformation at several instants are shown by the computer-generated film plots.

- 5045 ANALYSIS OF DYNAMICALLY LOADED NON-LINEAR STRUCTURES. Davidson, I. (9 Dale Lane, Appleton, Warrington, UK); Bradbury, J.N. (Nuclear Power Co. (Risley) Ltd. (UK)). pp E2/6 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

A CODES;CONTAINMENT SHELLS;CRACKS;
DEFORMATION;ELASTICITY;FBR TYPE REACTORS;
ITERATIVE METHODS;MECHANICAL STRUCTURES;
NONLINEAR PROBLEMS;P CODES;PRESTRESSED CONCRETE;
REACTOR SAFETY;SODIUM COOLED REACTORS;STRESSES

- 5046 ANALYSIS OF MECHANICAL HCDA CONSEQUENCES WITH ARES FOR DEMONSTRATION OF PRIMARY CONTAINMENT INTEGRITY IN SNR-300. Doerbecker, K.; Kellner, W.; Lauber, H. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp E1/5 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

A CODES;CONTAINMENT;DEFORMATION;FINITE DIFFERENCE METHOD;HYDRODYNAMICS;REACTOR ACCIDENTS;REACTOR CORES;SIMULATION;SNR-1 REACTOR;STRESSES;TWO-DIMENSIONAL CALCULATIONS

- 5047 TRANSIENT RESPONSE OF FAST-REACTOR CORE SUBASSEMBLIES TO PRESSURE LOADS: A COMBINED ANALYTICAL AND EXPERIMENTAL APPROACH. Donea, J.; Giuliani, S.; Holtbecker, H.; Verzeletti, G. (Commission of the European Communities, Ispra (Italy). Joint Research Centre). pp E3/6 1-9 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on

structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The intent of the paper is to describe a combined analytical-experimental programme that is being performed at J.R.C. Ispra in the field of subassembly deformation. From the numerical side, 2D and 3D non-linear dynamic finite element codes are being developed for coupled hydrodynamics structures situations. The structural part of these codes is based on the use of flexural elements in which non-linearities arising from elastic-plastic behaviour and from large-displacement effects are accounted for. As far as the fluid flow phase is concerned, a Lagrangian code is used, due to the particular configuration of the subassemblies, the fluid distortions can be assumed to be relatively small. Nevertheless with the analytical work experiments are performed in steps of increasing complexity so that each component of the computer codes can be adequately tested.

5048 ANALYSIS OF MECHANICAL HCDA CONSEQUENCES IN THE COOLANT LOOPS FOR DEMONSTRATION OF PRIMARY CONTAINMENT INTEGRITY IN SNR 300.

Giese, B.; Pollak, R. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp E4/5 1-9 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The demonstration of the accommodation of the mechanical consequences of a hypothetical core disruptive accident (HCDA) within the primary system is an essential objective in the safety assessment for SNR-300. For this purpose a mathematical model has been developed, in which the pressure pulse propagation in fluid-filled piping systems is calculated. This model takes into account the compressibility of the fluid as well as the elastic-plastic deformation of the pipe walls. It is the special aim of this work to estimate the attenuation of pressure peaks by elastic-plastic response of the pipe walls and to calculate the remaining strains. The basic system of partial differential equations includes the conservation laws for mass and momentum and the equations of state for the fluid and the pipe wall material. Additionally, the equation of motion for the pipe wall is provided, which considers the inertia of structure response after the pressure loading. The stress is described in a one-dimensional way. Axial stresses and strains are neglected. The numerical solution of the differential equations is carried out by an explicit extended Lax-Wendroff difference-scheme with the help of an intermediate step.

5049 LMFBR TYPICAL IMPULSIVE LOADING CHARACTERIZATION AND ITS APPLICATION TO SIMPLIFIED STRUCTURAL MODELS. Giuliano, V.; Lazzeri, L. (SAIGE S.p.A., Via P. Pesce 5, 1-16151 Genova-Sampierdarena, Italy). pp E2/7 1-16 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The aim of this paper is a discussion of the analytical methods which have been used for the dimensioning of the support structure of a LMFBR tank to cope with the explosive accident. Particular reference is given to fast but comprehensive methods developed in order to assist the designer in the preliminary stage of

the design. The three main steps necessary to define a general arrangement of the structure can be described as follows: (1) simplified equivalent characterization of the impulsive loading, (2) Characterization of the elastic plastic behaviour of simple structural elements, (3) parametric absorption of plastic energy with comparison with allowable strains. Each step has been extensively studied and the main results obtained in terms of the analytical methods developed and typical applications are presented and discussed. For step 1) a modified version of NONSAP code has been extensively used in order to analyze and to compare the effect of some different time histories of impulsive loading with that of a simplified 'equivalent' impulsive loading. For step 2) an original computer code (CARPA) has been developed and used for the elastic plastic analysis of beams of different sections. For step 3) a simple original computer code (CAT) based on an analytical integration technique, has been developed and used to evaluate parametric absorption of plastic energy correlated to the large displacements of the supported tank. Applications of these methods and codes are given.

5050 MAIN ASPECTS OF THE DESIGN OF A SUPPORT STRUCTURE OF A LMFBR WITH PARTICULAR REFERENCE TO THE EXPLOSIVE ACCIDENT CONSEQUENCES. Giuliano, V.; Lazzeri, L. (SAIGE S.p.A., Via P. Pesce 5, 1-16151 Genova-Sampierdarena, Italy). pp E2/8 1-15 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The aim of this paper is a review of the main aspects of the design of a support structure of a LMFBR tank, with particular reference to the analysis of the non-linear dynamic behaviour of the structure in the plastic range under the effect of an explosive accident within the tank. The structure is composed by a L-shaped flange, which supports the tank, connected by means of nine square beams to a rigid box-type ring, fixed to the concrete. The plug of the tank is connected to the L-shaped figure by means of a group of SS bars. The non-linear dynamic analysis of the explosive accident has been carried out on a lumped mass model, with elastic-plastic elements which simulate main components of the support structure and tank. The impulsive load connected to the explosive accident has been modelled (on the basis of extensive comparative studies carried out) as two triangular pressure impulses acting on the plug and on the bottom of the tank. A large amount of results, which describe displacements, velocities and accelerations of the plug, of the tank, and of the support structure, together with the forces and stresses acting on the main structural components are presented and discussed, with particular reference to the influence of the various parameters involved in the analysis.

5051 BULK COOLANT CAVITATION IN LMFBR CONTAINMENT LOADING FOLLOWING A WHOLE-CORE EXPLOSION. Jones, A.V. (Commission of the European Communities, Ispra (Italy). Joint Research Centre). pp E1/4 1-8 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

An LMFBR core undergoing an explosion

transmits energy to the containment in a series of pressure waves and the containment loading is determined by their cumulative effect. These pressure waves are modified by their interaction with the coolant through which they propagate. It is necessary to model both the induction of bulk cavitation by tension waves and the interaction of pressure waves with cavitating liquid in realistic containment loading calculations. This paper sets out the progress which has been achieved in such modelling and first indications for the effect of bulk coolant cavitation in LMFBR containment loading. Conclusions may be briefly summarised: 1) Bulk cavitation must be included in realistic containment loading calculations. 2) Phenomenological models of cavitating liquid without memory are inappropriate. The best approach is to model bubble dynamics directly, including at least momentum conservation and surface tension. 3) The containment loading resulting from a given explosion is sensitive to the state of preparation of the coolant. The number density of nucleation sites should therefore accompany the results of model tests.

- 5052 SUBASSEMBLY-TO-SUBASSEMBLY MELTTHROUGH ANALYTICAL STUDIES. Kulak, R.F.; Kennedy, J.M. (Argonne National Lab., IL (USA)). pp E3/2 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

BOILING;COMPUTER CALCULATIONS;CRACKS; DEFORMATION;DUCTS;FAILURES;FUEL ASSEMBLIES; LMFBR TYPE REACTORS;LOSS OF FLOW;MELTDOWN; NUMBER CODES;S CODES;THERMAL STRESSES

- 5053 FINITE-ELEMENT FORMULATIONS FOR THE THERMAL STRESS ANALYSIS OF TWO- AND THREE-DIMENSIONAL THIN REACTOR STRUCTURES. Kulak, R.F.; Kennedy, J.M. (Argonne National Lab., IL (USA)); Belytschko, T.B.; Schoeberle, D.F. (Illinois Univ., Chicago (USA)). pp E3/3 1-13 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper describes finite-element formulations for the thermal stress analysis of LMFBR structures. The first formulation is applicable to large displacement rotation problems in which the strains are small. For this formulation, a general temperature-dependent constituent relationship is derived from a Gibbs potential and a temperature dependent surface. A second formulation is presented for problems characterized by both large displacement-rotations and large strains. Here a set of large strain hypoelastic-plastic relationships are developed to linearly relate the rate of stress to the rate of deformation. These developments were incorporated into two ANL developed finite-element computer codes: the implicit version of STRAW and the 3D implicit Structural Analysis code. A set of problems is presented to validate both the 3D and 2D programs and to illustrate their applicability to a variety of problems.

- 5054 STRUCTURAL RESPONSE OF LARGE LMFBR HEAD CLOSURES TO HYPOTHETICAL CORE DISRUPTIVE ACCIDENTS. Kulak, R.F.; Marciniak, T.J. (Argonne National Lab., IL (USA)); Belytschko, T.B. (Illinois Univ., Chicago (USA)). pp E1/7 1-9 of Structural mechanics in reactor

technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

A finite-element model was developed to assess the ability of a pool-type LMFBR head closure to sustain an energetic HCDA. From a conceptual representation of an expanded EBR-II type deck. It was determined that there is a reasonable amount of symmetry in the structure that an initial model would consist of a 120° sector. This model accounts for the main structural elements of the closure head: beams, plates, inner ring, in-tank component nozzle and concrete shielding. The mass of the in-tank component was treated as a concentrated mass located at its attachment in the structural members. The ANL-developed SADCAT code was used in this study. The model was used to study the effect of several design variations on the structural response of the deck to an energetic HCDA. Two types of outer boundary conditions were considered to determine the effect on the closure head's displacement and stress levels. In addition to the above study, an analysis was conducted to determine the structural contribution of the concrete shielding.

- 5055 DETAILED NON-LINEAR ANALYSIS OF THE RESPONSE OF THE DECK OF A FAST BREEDER REACTOR UNDER ACCIDENTAL SODIUM IMPACT. Le Guern, M.; Dubois, J. (Engineering System International S.A., 20 rue saarinen, F-94150 Rungis Sillie, France). pp E1-6 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

EXPLOSIONS;FINITE ELEMENT METHOD;IMPACT SHOCK;NONLINEAR PROBLEMS;P CODES;REACTOR ACCIDENTS;REACTOR COMPONENTS;SODIUM;SUPER PHENIX REACTOR;THREE-DIMENSIONAL CALCULATIONS

- 5056 STRUCTURAL DYNAMICS STUDIES FOR SINGLE AND CLUSTERED SNR-300 FUEL ELEMENTS: A COMPARISON OF ANALYTICAL AND EXPERIMENTAL RESULTS. Liebe, R. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). pp E3/7 1-12 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Structural damage may occur in liquid metal cooled fast breeder reactors (LMFBR) when local failures escalate to a fast release of excessive thermal energy within one fuel element. Time scales (5 to 50. msec) and peaks of associated pressure pulses (50. to 500. bar) may exceed the burst pressure of the hexagonal wrapper containing the pin-bundle. Though after fuel rupture the pressure level in the vicinity is markedly reduced, still a significant pressure field is propagating through the core. Therefore surrounding subassemblies or control rods are exposed to impulsive differential pressures causing duct flexure as well as local cross section deformation. A theoretical and experimental research program at GfK Karlsruhe investigates the mechanical effects of conservative pressure transients on LMFBR fuel elements. As part of the program this paper describes two computer codes (BEDYN-2, CORE-1) for non-linear structural dynamics of single as well as clustered subassemblies.

- 5057 SIMULATION OF LMFBR EXCURSION MODELS BY MEANS OF ICECO. Marchertas, A.H.; Wang, C.Y.; Fistedis, S.H. (Argonne National Lab., IL (USA)). pp E2/5 1-11 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Some comparisons of transient fluid-structure tests with results from the ICECO containment code have already been published. Additional test data from the same series of experiments have again become available. Thus, the comparison of these more complex models with the ICECO results is the subject of this paper. The experimental data used herein pertain to test models composed of a spherical source within a cylindrical pool of water. The rigid ends of the container are connected by holddown bolts and cylindrical boundaries are made of one or two concentric deformable shells. The space above the surface of the water is occupied by air which may vent during the course of the experiment. In order to gain better insight into this problem, several analytical models with the thin vessel were studied. The principal means of comparison between experiment and analysis for both test configurations was the final deformation of the vessels. Certain pressure and impulse data at specified locations were also available and thus were used for comparison. The correspondence of experimental and analytical results was in general quite good.

- 5058 EXTENSION OF HCDA SAFETY ANALYSIS TO LARGE PCRV CONTAINMENT STRUCTURES. Marchertas, A.H.; Fistedis, S.H. (Argonne National Lab., IL (USA)); Bazant, A.P. (Northwestern Univ., Evanston, IL (USA)). Dept. of Civil Engineering; Belytschko, T. (Illinois Univ., Chicago (USA)). pp E4/1 1-12 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The prestressed concrete reactor vessel (PCRV) has thus far been applied for nuclear reactor containments involving steady-state loading. This paper, on the other hand, considers the PCRV for the LMFBR containment which must sustain dynamic effects stemming from the Hypothetical Core Disruptive Accident (HCDA). To perform a feasibility study of such a PCRV, a specialized formulation for transient analysis of such structures was developed. The analytical model of a PCRV is represented in a finite element computer code involving an explicit time-integration procedure. The formulation is particularized for the axisymmetric geometry and includes simulations of tensile cracking of concrete, reinforcement and prestressing capability. The code is applied to the analysis of a PCRV in a possible LMFBR configuration. The code provides a useful tool for evaluating the effect of reinforcement and prestressing on crack propagation. By means of the code, the design parameters can be varied to improve and optimize the PCRV configuration.

- 5059 ENERGY SOURCE FOR MODELING HCDA BUBBLES. Ploeger, D.W.; Cagliostro, D.J.; Goodale, T.C. (Stanford Research Inst., Menlo Park, CA (USA)). Poulter Labs.). pp E4/2 1-12 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam,

Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

One concern in the safety analysis of liquid metal fast breeder reactor (LMFBRs) is the release of radioactive core materials to the environment following a hypothetical core disruptive accident (HCDA). The release depends on the transport properties of the expanding, two-phase, fuel-coolant bubble produced during the HCDA and the energy available in the bubble to damage the reactor containment system. Most predictions of the HCDA bubble pressure-volume change relationship assume an isentropic expansion of the bubble material from some known initial state. This work describes the development and characterization of an energy source that models the assumed initial conditions and the subsequent expansion of an HCDA bubble in a simple 1/30-scale model of the Clinch River Breeder Reactor (CRBR). The source is used in a study of the phenomena associated with expanding, rising HCDA bubbles, the transport of radioactive core materials to the cover gas region, and the loading resulting from the impact of the coolant slug with the vessel cover. The results imply that gaswork and slug kinetic energy predictions based on an equilibrium expansion of a superheated liquid may be too high by a factor of two.

- 5060 ANALYSIS OF HYDRODYNAMIC EFFECTS OF LARGE SODIUM-WATER REACTIONS. Sakano, K.; Shindo, Y.; Koishikawa, A.; Maekawa, I. (Nuclear Power Engineering Department, Technology Division, Kawasaki Heavy Industries, Ltd, 4-1, 2-Chome, Hamamatsu-cho, Minato-ku, Tokyo, 105, Japan). pp E5/5 1-12 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Large leak sodium-water reactions that would occur in a steam generator of LMFBR causes abrupt changes of pressure and velocity of fluid in a secondary sodium system and relief system. These changes exert unsteady forces on piping and components. The estimation of the effect of the pressure and forces is needed to assess the integrity of the system. SOWACS-III code has been developed to predict the pressure, velocity, flow-induced forces and the response of the piping and components to these forces. This paper described SOWACS-III together with its model and method. Results of analyses are also given, the comparison with experimental results of initial pressure spike being included.

- 5061 CRBRP STEAM GENERATOR MODELING TECHNIQUES FOR PRESSURE EXCURSIONS GENERATED BY SODIUM/WATER REACTIONS. Shimozone, G.S. (Atoms International Div., Canoga Park, CA (USA)). pp E5/8 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

CLINCH RIVER BREEDER REACTOR; LEAKS; MOLTEN METAL-WATER REACTIONS; PIPES; SIMULATION; SODIUM; STEAM GENERATORS; SUPERHEATERS; T CODES

- 5062 ANALYSIS OF EBR-II STEAM-GENERATOR SYSTEM FOR PRESSURE PULSES DUE TO LEAKS OF STEAM OR WATER INTO SODIUM. Srinivas, S.;

- Chopra, P.S. (Argonne National Lab., IL (USA)). pp E5/4 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
- From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
- EBR-2 REACTOR;EVAPORATORS;LEAKS;LMFBR TYPE REACTORS;MOLTEN METAL-WATER REACTIONS;SECONDARY COOLANT CIRCUITS;SODIUM;STEAM GENERATORS; SUPERHEATERS;TRANSIENTS
- 5063 PARAMETRIC STUDY OF THE NONLINEAR BEHAVIOUR OF COMPLETE LMFBR SUBASSEMBLY USING THE CORTAN COMPUTER CODE. Zehlein, H. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). pp E3/8 1-13 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
- From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
- The paper emphasizes the computational aspects of combined analytical-experimental investigations concerning dynamic elastoplastic deformation of a single fuel subassembly under transverse pressure loading. To simulate the situation within a fast reactor core during a postulated local vapor explosion the cushioning effect of the thin sodium layers between subassemblies has to be included. Therefore a new computer code CORTAN was developed which combines both a one-dimensional axial squeeze flow model as well as a variable cross-section Timoshenko beam model. The ensuing program structure of the code is described.
- 5064 COMPUTER SIMULATION OF LMFBR PIPING SYSTEMS. A-Moneim, M.T.; Chang, Y.W.; Flatedis, S.H. (Argonne National Lab., IL (USA)). pp E4/3 1-14 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
- From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
- A two-dimensional coupled hydrodynamic-structural response analysis of piping systems is described. Implicit Continuous-Fluid Eulerian (ICE) technique is utilized in the hydrodynamics while a finite-element technique is used in the structural analysis. Different piping components such as elbow, valves, reducers, expansions, heat exchangers, and tees are modelled and coupled with the straight pipe model. An axisymmetric general component model that can be used in modelling valves, reducers, expansions, and heat exchangers is described. At the inlet and outlet region of such component the cross-sectional area may change suddenly or gradually, or may not change at all. Among the options available in this model are deformable exterior walls, interior rigid wall simulation, and tube bundle effect. Sample problems are presented to demonstrate the capabilities of the method; their results and future developments are discussed.
- 5065 (ANL-Trans--1173) LARGE SCALE PRIMARY SODIUM PUMPS. Guidaz, J.; Canavelis, R.; Guilloiseau, P. Translated from Houille Blanche; No. 7/8, 719-725(1977). 17p. Dep. NTIS, PC A02/MF A01.
- A design study on primary sodium pumps for fast reactors is described. The study was performed for a 1200 MW(e) reactor with 4 pumps.
- 5066 OPTIMIZATION OF A FAST HIGH-POWER REACTOR OF LMFBR TYPE BY MEANS OF A NON-CONVEX PROGRAMMING MATHEMATICAL MODEL. Pavelescu, M.; Dumitrescu, H. (Institutul de Reactori Nucleari Energetici, Pitesti (Romania)); Ochiana, G. (Institutul de Fizica si Inginerie Nucleara, Bucharest (Romania)). Rev. Roum. Phys.; 22: No. 9, 997-1003(1977).
- The problem of minimizing the doubling time for an LMFBR is considered. The decision vector is a distribution one of fissile and fertile material in the reactor which is considered a static system in initial state, and the mathematical model for optimization is a non-convex programming model. Modelling of the optimization problem is carried out by the SUMT method.
- 5067 (ANL-Trans--1174) PRIMARY SODIUM PUMPS OF PHENIX AND SUPERPHENIX. Chabassier, R.; Blateyron, J.; Guiton, P.; Canavelis, R.; Roumailhac, J.; Gravier, M. Translated from Houille Blanche; No. 7/8, 709-716(1977). 20p. Dep. NTIS, PC A02/MF A01.
- This report shows how design experience at Hispano-Suiza (SNECMA Division), Bergeron SA and the French Atomic Energy Commission has determined the hydraulic and mechanical features of the primary sodium pumps for the Phenix and Superphenix reactors.
- 5068 MODELING OF LMFBR FUEL CLADDING MECHANICAL BEHAVIOR UNDER TRANSIENT CONDITIONS. Bard, F.E. (Hanford Engineering Development Lab., Richland, WA (USA)). pp C4/3 1-12 of Structural mechanics in reactor technology. Vol. C. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
- From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
- Some aspects of continuum physics used to develop the new constitutive equations for fuel cladding are presented in this report. The theory of internal state variable, whose evolution is governed by ordinary differential equations, has been introduced into continuum mechanics. The components of the internal or hidden state variable vector may be scalars, components of vectors, components of second order tensors, etc. Irradiation, hardening, grain size, chemical reaction, etc., are scalar internal variables. Internal strain, dislocation loop density, etc., are all examples of second order tensors whose components are internal state variables. The internal state variables of this report are irradiation, ϕ , which satisfies the neutron density conservation law, and hardening. The isotropic hardening plasticity theory used in conjunction with creep has been incorporated into a computer code called PECTCLAD and has been correlated to annealing tests of 20X cold-worked Type 316 stainless steel and to tensile tests of annealed Type 316 stainless steel cladding. This correlated model is used to accurately predict the tensile, burst, and Fuel-Cladding Transient Testing (FCTT) data for 20X cold-worked Type 316 stainless steel cladding over a wide range of temperatures, temperature rates and strain rate. An excellent comparison is obtained between the predicted and measured data.
- 5069 SIMPLE STEEL/WATER MODEL FOR PRELIMINARY STUDIES OF ACOUSTIC VIBRATION IN LMFBR. Bentley, P.G.; Firth, D.; Rowley, R.; Beesley, M. (UKAEA, Risley. Nuclear Power Development Establishment). pp F5/7 1-11 of Structural

mechanics in reactor technology. Vol. F. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

One source of vibration excitation in Liquid Metal Fast Breeder Reactor is the acoustic energy which is generated by the circulating pump and transmitted through the fluid to various structural components. To gain some preliminary understanding of the extent and magnitude of vibration in fast reactors therefore, a simple model has been constructed in which only the major components are represented. The modeling theory is discussed and it is shown that adequate representation of the steel/sodium reactor materials can be obtained in the model based on the use of steel/water. Appreciable stresses are found on all the components in the primary circuit, not necessarily only those close to the simulated pump source. After scaling them to reactor size and allowing for the source calibration, it is found that stresses are unlikely to be sufficiently high to cause damage. The possibility of higher stresses being produced by better acoustic/structure coupling, similar to that found in smaller laboratory experiments, is to be further investigated on the model.

- 5070 THERMAL AND STRESS ANALYSES OF MELTDOWN CUPS FOR LMFBR SAFETY EXPERIMENTS USING SLSF IN-REACTOR LOOPS. Blomquist, C.A.; Pierce, R.D.; Pedersen, D.R. (Argonne National Lab., IL (USA)); Ariman, T. (Notre Dame Univ., IN (USA)). pp E4/4 1-11 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The test trains for the Sodium Loop Safety Facility (SLSF) in-reactor experiments, which simulate hypothetical LMFBR accidents, have a meltdown cup to protect the primary containment from the effects of molten materials. Thermal and stress analyses were performed on the cup which is designed to contain 3.6 kg of molten fuel and 2.4 kg of molten steel. Thermal analyses were performed with the Argonne-modified version of the general heat transfer code THTB, based on the instantaneous addition of 3200 °K molten fuel with a decay heat of 9 W/gm and 1920 °K molten steel. These analyses have shown that the cup will adequately cool the molten materials. The stress analysis showed that the Inconel vessel would not fail from the pressure loading, it was also shown that brittle fracture of the tungsten liner from thermal gradients is unlikely. Therefore, the melt-down cup meets the structural design requirements.

- 5071 SUBASSEMBLY BOWING EXPERIENCE IN EBR-II. Bottcher, J.H.; Hofman, G.L. (Argonne National Lab., IL (USA)). pp D2/9 1-7 of Structural mechanics in reactor technology. Vol. D. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Bowing was not considered a problem in EBR-II until subassembly handling problems occurred. A typical handling problem, caused by top end deflection of the bowed subassembly, is centering of the subassembly removal mechanism over the subassembly. Another problem is excess loads of up to 375 kg that have been required

to remove bowed subassemblies from their grid position. All of the bowed type 304 stainless steel subassemblies that created handling difficulties were given a post-irradiation examination. Correlation of the in-core bowing phenomena with the bowing data base required both an axial and radial thermal and flux characterization, consideration of the subassembly design, interpretation of swelling and creep behavior and a simple workable model that could incorporate these factors. A thermal hydraulics code, 'CLUSTER' was used to calculate the operating temperatures at sixty radial locations for each of nineteen axial sections along the hex duct. For this analysis a simple beam method was used with postulated radial restraint bounds in simulating intersubassembly loading. The results of this analysis showed that the bow at a neutron fluence of $>2.7 \times 10^{22} \text{ n/cm}^2$, $E>0.1 \text{ MeV}$, in these structurally varying subassemblies was highly thermal gradient dependent.

- 5072 FROM PHENIX TO SUPER PHENIX: MECHANICAL STRUCTURES ASSURING REACTOR VESSEL TIGHTNESS AT MAIN SODIUM PUMP PENETRATIONS. Chabassier, R.M.; Blateyrou, J.J. (Societe Nationale d'Etude et de Construction de Moteurs d'Aviation (SNECMA), 92 - Suresnes (France). Div. Hispano-Suiza; Roumiliac, J.F. (Compagnie d'Ingenierie pour les Reacteurs au Sodium (CIRNA), Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)). pp F2/6 1-12 of Structural mechanics in reactor technology. Vol. F. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

It is shown how the mechanical problems concerning the tightness of main sodium pump penetrations through the deck of a pool type reactor have been solved in joint SNECMA - HISPANO-SUIZA/CIRNA studies. Extensive general studies have been performed and a comprehensive dossier of different possible solutions compiled. Wherever appropriate, information has been extrapolated from PHENIX experience. In the solution finally adopted, an elastic torus shaped ring, which forms a ball-and-socket type joint is used. A thin shell structure connects the bottom to the reactor internals in such a way as to permit both vertical and rotational displacements. Studies and calculation have been completed with the realization of a 1/5 scale model of the upper torus ring. Displacement, and angular and axial stiffness measurements have confirmed the validity of the hypotheses.

- 5073 PARAMETRIC INVESTIGATION OF DYNAMIC FRACTURE. Chopra, P.S.; Moustakakis, B. (Argonne National Lab., IL (USA)). pp D3/6 1-10 of Structural mechanics in reactor technology. Vol. D. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper presents results of parametric finite element fracture mechanics analyses conducted to develop an analytical model for simulating dynamic brittle fracture. Although this work may have wider general application, the dynamic loads considered here are those that may arise due to rapid release of fission gases from a failed fuel element inside a subassembly duct in an LMFBR. The analyses were conducted on an edge-cracked panel with various configurations of primary and secondary flaws

similar to those investigated for quasi-static fracture earlier by Chopra, et al. For the fracture configurations analyzed, the peak dynamic crack opening stresses (COS) in the vicinity of the crack tip exceeded the static values by a factor of approximately 1.5. The displacement between two opposing nodes behind the crack tip, called the crack opening displacement (COD), also was investigated. The dynamic COD was found to be approximately 1.3 times greater than the corresponding static value.

- 5074 TEMPERATURE AND STRESS RESPONSES OF CLEAN EVAPORATOR TUBES UNDER OSCILLATORY WATER RIVULET ACTION IN DNB REGION. Chu, C.L.; Roberts, J.M.; Dalcher, A.W. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). pp F9/3 of Structural mechanics in reactor technology. Vol. F. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

DEPARTURE NUCLEATE BOILING;EVAPORATION; FINITE ELEMENT METHOD;LMFBR TYPE REACTORS; RESPONSE FUNCTIONS;STEAM LINES;STRESS ANALYSIS; THERMAL ANALYSIS;TUBES;WATER;EVAPORATORS

- 5075 TWO-DIMENSIONAL HEAT CONDUCTION CALCULATIONS ON THE PIN CLADDING AND COOLANT OF A WIRE WRAP BUNDLE. Cornet, G. (Societe Belge pour l'Industrie Nucleaire, Brussels). pp D3/3 of Structural mechanics in reactor technology. Vol. D. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

BREEDING BLANKETS;F CODES;FINITE ELEMENT METHOD;FUEL CANS;FUEL ELEMENT CLUSTERS;LMFBR TYPE REACTORS;NUSSELT NUMBER;SPACERS;STEADY-STATE CONDITIONS;THERMAL CONDUCTION;TWO-DIMENSIONAL CALCULATIONS;WIRES;HEAT TRANSFER

- 5076 LMFBR FUEL ASSEMBLY CHANNEL WALL CROSS SECTION OPTIMIZATION. Dubberley, A.E.; Johnson, D.P.; Patches, J.R.; McCandless, R.J. (General Electric Co., Sunnyvale, CA (USA)). Fast Breeder Reactor Dept.). pp D2/2 1-9 of Structural mechanics in reactor technology. Vol. D. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The problem addressed in this paper is the optimal design of an LMFBR fuel assembly channel, subject to realistic material properties, operating environment and reactor geometric limitations. An LMFBR fuel assembly channel is a hexagonal thin-walled tube surrounding the fuel pin bundle to facilitate fuel handling and to control the flow of coolant through the core. The environment results in a ballooning of the channel from its original hexagonal shape due to irradiation swelling, thermal and irradiation enhanced creep, and the elastic pressure deformation of the channel material. The cross-section shape of the hexagonal channel is limited primarily by the principal stress and the total channel radial growth. It is presently considered preferable that adjacent channels do not touch from channel ballooning due to concerns about safety and refueling considerations. A computer program, WALLOP, has been developed as described in this paper to derive the wall shape of such an optimum channel, given

realistic design and material constraints. Within WALLOP, an initial cross-sectional shape is chosen and a beam model of the wall is used to determine the elastic deflection. The computer program sequentially perturbs the wall cross section and repeats the process until channel life is maximized. For prototypic 1200 MWe LMFBR cores, comparing uniform thickness channel walls and optimized cross section walls generally shows that wall optimization yields about a 25% to 50% increase in allowable channel life for the same amount of material.

- 5077 STRESS-STRAIN PROPERTIES OF FAST REACTOR FUEL CLADDING UNDER CONSTANT PRESSURE TRANSIENT HEATING CONDITIONS. Hunter, C.W.; Johnson, G.D. (Hanford Engineering Development Lab., Richland, WA (USA)). pp C4/7 of Structural mechanics in reactor technology. Vol. C. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

DEFORMATION;DUCTILITY;FRACTURE PROPERTIES; FUEL CANS;HIGH TEMPERATURE;LMFBR TYPE REACTORS; LOSS OF FLOW;MICROSTRUCTURE;STAINLESS STEEL-316; STRAINS;STRESSES;TRANSIENTS;VERY HIGH TEMPERATURE;STRESS ANALYSIS

- 5078 DEVELOPMENT OF A FINITE-STRAIN FUEL-ELEMENT MODEL. Jankus, V.Z.; Shack, W.J. (Argonne National Lab., IL (USA)). pp D1/4 1-9 of Structural mechanics in reactor technology. Vol. D. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The LIFE-III computer code has been developed to describe the steady-state performance of LMFBR fuel rods. The mechanical analysis in the original version of the code was based on the assumption that the strains in the fuel and the cladding could be considered as infinitesimal. However, this assumption is grossly violated in the vicinity of the central void. To provide a more accurate description of the mechanical behaviour of the fuel element, a finite-strain fuel-element model was developed. The finite-strain calculations in the modified code are based on an incremental analysis. The deformations are assumed to be axisymmetric. The fuel element is divided into a number (≤ 10) of axial segments. Each segment is assumed to be in a state of generalized plane strain. The deformation history of the fuel element is analyzed in discrete time steps, and the basic geometry of the element is updated after each time step. To carry out the calculation for nonlinear materials, each axial segment is divided into a number (≤ 25) of concentric cylindrical rings. The analysis maintains the overall force equilibrium of each ring and enforces geometric compatibility between the rings. To test the adequacy of the formulation, the code was applied to a number of benchmark problems, and excellent results were obtained. Sample calculations with actual fuel pins show that the modified code gives more satisfactory results in the fuel regions near the central void. The finite-strain capability requires only a modest increase (approximately 5%) in computing time.

- 5079 VIBRATIONAL ANALYSIS OF THE SUPER PHENIX INTERNAL SHELLS. Jeanpierre, F.; Livolant, M. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)). Dept. des Etudes

- Mecaniques et Thermiques). pp F5/5 of Structural mechanics in reactor technology. Vol. F. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
FLUCTUATIONS; FLUID FLOW; LMFBR TYPE REACTORS; MECHANICAL VIBRATIONS; REACTOR COMPONENTS; RESONANCE; SHELLS; STRAINS; SUPER PHENIX REACTOR
- 5080 THERMAL/HYDRAULIC BOWING STABILITY ANALYSIS OF GRID-SUPPORTED MULTI-PIN BUNDLES WITH DIFFERENTIAL SWELLING AND IRRADIATION CREEP. McAreevey, G. (UKAEA, Risley, Nuclear Power Development Establishment). pp D2/1 1-15 of Structural mechanics in reactor technology. Vol. D. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
Azimuthal variations of clad temperature in fuel pin bundles leads to pin bowing by differential thermal expansion. During irradiation in a fast flux further possibly more severe bowing is caused by differential neutron induced voidage swelling, which, being temperature sensitive, will also vary azimuthally. The problem of pin bowing in a fuel element cluster involves consideration of the thermal/hydraulic behaviour, allowing for both inherent and induced clad temperature non-uniformities, coupled with the restrained bowing behaviour, including differential thermal expansion, differential swelling, and irradiation creep. All pins must be considered simultaneously. In the temperature and stress ranges of interest thermal creep may be neglected. An existing computer code, IAMBIC solves the zero time thermal bowing problem for a cluster of up to 61 pins on hexagonal pitch, with up to 21 supports at arbitrary axial spacing. The present paper describes the basis of TRIAMBIC, a time dependent code which analyses the irradiation induced effects in fuel pin bundles due to fast neutrons.
- 5081 THREE-DIMENSIONAL ANALYSIS FOR FUEL PIN DEFORMATIONS IN AN LMFBR ASSEMBLY. Miki, K. (Hitachi Ltd., Kawasaki, Kanagawa (Japan). Atomic Energy Research Lab.). pp D2/3 1-16 of Structural mechanics in reactor technology. Vol. D. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
A new analytical method is presented for analyzing, in three dimensions, the mechanical response of fuel pins with wire spacers, to their thermal and neutronic environment in an assembly of a Liquid Metal cooled Fast Breeder Reactor (LMFBR). It analyzes the mechanical interactions of the fuel pins in the assembly corresponding to three directions, which form an angle of $\pi/3$ radians with one another, and is based on the mathematical relationship of the displacement at the contact points and the associated contact forces with respect to all fuel pins in a direction corresponding to one of three axes. The displacement of the fuel pin is composed of its three components in each of the three directions thus calculated. In particular this method makes it possible to take into precise consideration the two factors affecting the deformation of the fuel pins - (a) the mutual interaction of fuel pins and (b) the forced breeding of the fuel pins due to the deformation of the wrapper tube. Based on this method, a new computational code, the Subchannel Deformation Analysis Code for Wire-Wrapped Assemblies (SHADOW) has been developed which can be applied to seven kinds of wire-wrapped fuel assemblies with a number of fuel pins ranging from 7 to 169.
- 5082 COMPUTERIZED THERMAL INPUT FOR STEAM GENERATOR STRESS ANALYSIS. Moody, E.; Rico, L.G.; Skopp, G.H. (Atomica International Div., Canoga Park, CA (USA)). pp F9/5 1-11 of Structural mechanics in reactor technology. Vol. F. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
In order to perform structural analyses of the CRBRP steam generators which include the effects of various system thermal hydraulic transients, detailed thermal profiles must be developed and transmitted to the stress analyst in a usable form. A unique, semi-automated computer system has been developed to optimally accomplish this task. This method has enabled very rapid analyses of complex thermal stress problems. For scoping purposes, simplified models have been used to determine the worst transients thus minimizing detailed analysis. The method has also permitted rapid analysis of the effect of design changes. While the method was developed for steam generator analysis, it is being applied, in full or part, to thermal stress analysis of other components resulting in significant engineering cost and time savings.
- 5083 FLOW-INDUCED VIBRATION BENCH TESTING OF REACTOR COMPONENTS. Mulcahy, T.M. (Argonne National Lab., IL (USA)). pp F6/8 of Structural mechanics in reactor technology. Vol. F. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
CLINCH RIVER BREEDER REACTOR; FLUID FLOW; LMFBR TYPE REACTORS; MECHANICAL VIBRATIONS; REACTOR COMPONENTS; SIMULATION; TURBULENCE; VORTICES
- 5084 LMFBR-FUEL CLADDING THERMAL CREEP STRAIN UNDER LINEARLY INCREASING INTERNAL PRESSURE. Nakatsuka, M. (Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan). Research and Development Center). pp C4/8 1-11 of Structural mechanics in reactor technology. Vol. C. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
This paper describes an apparatus designed to simulate the continuously increasing internal pressure due to the accumulation of fission product gas. There is also an examination of the effects of increasing pressure on the thermal creep rate of cladding. Test specimens are 20% Cu type 316 stainless steel tubes with 6.5 mm OD, 5.6 mm ID and 200 mm length. Increasing pressure creep tests were performed at 700°C, the hoop stress was increased in steps with 3 incremental rates of 0.15, 0.075 and 0.0375 kg/mm²h. Constant pressure creep tests were also performed at the same temperature for comparison. The results show that the creep rates obtained from creep curves under increasing pressure are several times as high as the stationary creep rates measured under the same constant pressure. Creep strain rate under increasing pressure, due to the accumulation of gaseous fission

products in the actual fuel cladding, is approximately 2 to 5 times that of the constant pressure creep rates, although the creep rate magnification depends on the stress and the stress-increase rate.

- 5085 RESPONSE OF A GRID SPACED FUEL ASSEMBLY TO SHIPPING ENVIRONMENT. Pal, N.; Allen, D.C. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). pp D2/12 1-12 of Structural mechanics in reactor technology. Vol. D. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

This paper presents a dynamic analysis approach for design verification of a grid spaced nuclear fuel under conditions of shipping and handling using prototypical test examples of transportation by truck. Analytical response results were obtained using representative trailer bed acceleration time histories which had been recorded earlier under simulated shipping by truck transport environment. Important parameters affecting response are identified. Recognizing that the analytical estimate of the fundamental periods of the GSA cannot be very accurate owing to the complex geometry, upper bounds on acceleration responses were established using the standard response spectrum technique. Finally, the GSA design was verified by laboratory tests—both static and dynamic. Static tests were used to calibrate displacement response of the grid spacer tabs relative to the fuel rods. Prototypical dynamic shaking table tests were next employed in which the response of the GSA was determined in sinusoidal sweep type of excitation. Handling loads were represented by shock tests consisting of one-cycle impulses at various levels of acceleration. The results of laboratory testing were seen to reinforce credence of design verification by analysis of the GSA subjected to shipping and handling environment.

- 5086 PROBABILISTIC DESIGN METHOD FOR LMFBR FUEL RODS. Peck, S.O.; Lovejoy, W.S. (General Electric Co., Sunnyvale, CA (USA). Fast Breeder Reactor Dept.). pp C4/9 1-8 of Structural mechanics in reactor technology. Vol. C. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Fuel rod performance analyses for design purposes are dependent upon material properties, dimensions, and loads that are statistical in nature. Probabilistic design methods provide a systematic way to reflect the uncertainties in design parameters. PECS-III is a computer code which employs Monte Carlo techniques to generate the probability density and distribution functions for time-to-failure and cumulative damage for sealed plenum LMFBR fuel rods on a single rod or whole core basis. The probability that some measure of rod performance will fall within given limits is estimated by the relative frequency with which the Monte Carlo samples fall within those limits. The PECS-III deterministic model is a one-dimensional structural analysis code. Cladding loading mechanisms considered include fission gas pressure, differential fuel-cladding thermal expansion, and differential fuel-cladding swelling. The model predicts thermal creep strain, irradiation creep strain, irradiation induced swelling, and cumulative damage fractions based on strain and time to rupture. In fuel rod design, it is oftentimes

low probability events (e.g. failure) that are of interest. Importance sampling techniques are used in PECS-III to accurately reproduce the output distribution in the area of interest without the high cost of a large number of samples.

- 5087 STRESS-INTENSITY FACTORS FOR IRRADIATION-EMBRITTLED HEXAGONAL SUBASSEMBLY DUCTS. Petroski, H.J.; Glazik, J.L. (Argonne National Lab., IL (USA)); Achenbach, J.D. (Northwestern Univ., Evanston, IL (USA). Technological Inst.). pp D2/11 1-7 of Structural mechanics in reactor technology. Vol. D. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In the present work, the results of a finite-element analysis of the problem of a crack in a corner of a uniformly pressurized hexcan are reported, and they support the applicability of beam theory to the hexcan problem. In particular, modeling the hexcan corner as a cracked strip whose crack faces are loaded with the hexcan stress distribution has been found to give very good results for the particular hexcan studied. The finite-element results for uniformly pressurized hexcans have been used as the basis for computing stress intensity factors for cracks in non-uniformly loaded hexcans, under loading conditions that might simulate those of fission gas released from failed fuel pins or other non-uniform accident loading conditions. A finite-element analysis also has been carried out for the cracked hexcan loaded by midflat forces. The results of the different techniques are compared, and the simple model of a cracked strip loaded with the hexcan stress distribution is found again to agree closely with the finite-element results for the hexcan considered. However, the weight function technique is expected to apply to geometries for which the strip model is inadequate and to hold over a wider range of crack sizes.

- 5088 STATISTICAL MECHANICAL ANALYSIS OF LMFBR FUEL CLADDING TUBES. Poncelet, J.P.; Pay, A. (Societe Belge pour l'Industrie Nucleaire, Brussels). pp C4/2 1-14 of Structural mechanics in reactor technology. Vol. C. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The most important design requirement on fuel pin cladding for LMFBR's is its mechanical integrity. Disruptive factors include internal pressure from mixed oxide fuel fission gas release, thermal stresses and high temperature creep, neutron-induced differential void-swelling as a source of stress in the cladding and irradiation creep of stainless steel material, corrosion by fission products. Under irradiation these load-restraining mechanisms are accentuated by stainless steel embrittlement and strength alterations. To account for the numerous uncertainties involved in the analysis by theoretical models and computer codes statistical tools are unavoidably requested, i.e. Monte Carlo simulation methods. Thanks to these techniques, uncertainties in nominal characteristics, material properties and environmental conditions can be linked up in a correct way and used for a more accurate conceptual design.

- 5089 EXPERIMENTS ON THE RESPONSE OF HEXAGONAL

SUBASSEMBLY DUCTS TO INTERNAL PRESSURE PULSES. Romander, C.M.; Cagliostro, D.J. (Stanford Research Inst., Menlo Park, CA (USA)). Poulter Labs.). pp E3/4 1-9 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The safety of Liquid Metal Fast Breeder Reactors depends in part on the ability of the hexagonal subassembly ducts (hexcans) to withstand accidental internal radial loads so that damaged hexcans can be removed from the reactor core. To predict deformations and strains in hexcans subjected to these internal loads and in hexcans surrounding the damaged hexcan, Argonne National Laboratory has developed the finite-element, two-dimensional structural code STRAW (Structural Transient Response of Assembly Wrappers). The experiments described here were performed to verify the modeling techniques used in STRAW and to determine the effects of material properties on hexcan response. Pressure pulses were applied in 12-inch-long hexcans of annealed, 20% CW, and 50% CW 316 stainless steel. The results of these experiments are presented in a form suitable for verifying two-dimensional structure codes designed to predict the response of hexagonal subassembly ducts to internal pressures.

5090 STRUCTURAL ANALYSIS OF EXPERIMENTAL CARBIDE FUELED DRIVER ASSEMBLY FLOW DUCT FOR TESTING IN THE FFTF. Rowe, G.H.; Mehta, D.D.; Desaulniers, W.E. (Combustion Engineering, Inc., Windsor, CT (USA)). pp D2/10 1-14 of Structural mechanics in reactor technology. Vol. D. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Mixed carbide fueled driver assembly experiments will be tested in FFTF fuel driver positions as part of the National Advanced Fuel Program. The design of the experiment flow ducts must assure conformance to FFTF functional requirements in addition to service as a test vehicle for the carbide fuel irradiations. Test goals of damage fluence burnup, and fluence to burnup ratio exceed those of the standard oxide fueled drivers. As a consequence, the 20% cold worked type 316 stainless steel material of construction will experience significant irradiation induced creep and swelling. Additionally, the flow duct design must withstand the enhanced thermal transients produced by the action of carbide fuel during reactor scrams. A major FFTF functional requirement is that adjacent flow ducts do not touch each other except at the load pads. This requires a realistic analysis of the creep and swelling deformation of the flow duct during an experiment lifetime. Flow duct dilation, i.e., diametrical growth of the hexagonal cross section, was investigated by the MARC-CDC finite element program. Subroutines for irradiation creep and swelling were incorporated into the program.

5091 PROOF TESTS OF IRRADIATED EBR-II DUCTS. Ruther, W.E.; Staffou, J.D.; Chopra, P.S.; Lambert, J.D.B.; Srinivas, S. (Argonne National Lab., IL (USA)). pp D3/7 1-8 of Structural mechanics in reactor technology. Vol. D. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San

Francisco, CA, USA (15 Aug 1977).

Permanent deformation and fracture of subassembly ducts due to potential pressure pulses generated by the discharge of fission gas from fuel elements are important considerations in determining the burnup limit of fuels in LMFBR's. This paper discusses the results of a series of dynamic pressure tests that were conducted on intentionally defected irradiated EBR-II subassembly ducts. The tests were designed to simulate failure of a fuel element in a cluster of such elements. The major objective of the tests was to provide complete assurance that cladding failure of a driven-fuel element, which might conceivably cause rapid release of stored fission gas, would not deform or otherwise damage subassembly ducts in a way that would hinder movement of a control rod. The test program described provides a reliable method of assessing the failsafety of LMFBR ducts. The test procedure is particularly important since present analytical solutions cannot adequately model the subassembly and the reactor conditions.

5092 INITIAL PRESSURE SPIKE AND ITS PROPAGATION PHENOMENA IN SODIUM-WATER REACTION TESTS FOR MONJU STEAM GENERATORS. Sato, M.; Hiroi, H.; Tanaka, N.; Hori, M. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). pp E5/6 1-12 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

With the objective of demonstrating the safe design of steam generators for prototype LMFBR MONJU against the postulated large-leak accident, a number of large-leak sodium-water reaction tests have been conducted using the SWAT-1 and SWAT-3 rigs. Investigation of the potential effects of pressure load on the system is one of the major concerns in these tests. This paper reports the behaviour of initial pressure spike in the reaction vessel, its propagation phenomena to the simulated secondary cooling system, and the comparisons with the computer code for one-dimensional pressure wave propagation problems.

5093 EFFECT OF TUBE-SUPPORT INTERACTION ON THE DYNAMIC RESPONSE OF HEAT EXCHANGER TUBES. Shin, Y.S.; Jendrzyszczak, J.A.; Wambganuss, M.W. (Argonne National Lab., IL (USA)). pp F6/5 1-11 of Structural mechanics in reactor technology. Vol. F. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

To avoid detrimental tube vibration in heat exchangers, resonant conditions and instabilities must be avoided, and/or peak dynamic amplitudes must not exceed allowable limits. In attempting a theoretical analysis, questions arise as to the effects of tube/support interaction on tube vibrational characteristics (i.e. resonant frequencies, modes, damping) and response amplitude. As a part of ANL's Flow-Induced Vibration Program in support of the Clinch River Breeder Reactor Plant (CRBRP) steam generator design activity, tube/support interaction experiments are being performed not only to gain the insight into the dynamic behavior of CRBRP steam generator tubes, but also to provide the basis for developing design guidance. Test results were compared with analytical results based on

multispan tube with 'knife-edge' supports at the support locations.

- 5094 MODEL FOR PRESSURE IN AN LMFBR DUCT DUE TO DISCHARGE OF GAS FROM A FAILED FUEL PIN. Srinivas, S.; Chopra, P.S. (Argonne National Lab., IL (USA)). pp D3/5 1-12 of Structural mechanics in reactor technology. Vol. D. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In this paper an analytical model for the calculation of pressure pulses in hexagonal ducts due to discharge of gas from a failed fuel pin is developed. The analysis yields the time history of the pressure pulse which can be used in the calculation of permanent deformation of the duct or in the assessment of the susceptibility of the duct to fracture. The real physical situation of gas discharging through a pin in a duct filled with liquid is complex to model. Here the phenomenon is modeled based on some reasonable assumptions. In this model the analysis is divided into two stages. In the first stage the gas expands as a spherical bubble, but the influence of the duct wall is taken into account. At the end of the first stage the spherical shape of the bubble is assumed to be lost and the gas is assumed to expand axially as a column. The analysis involves solving the continuity and momentum equations for the liquid along with the energy balance equation for the gas.

- 5095 ANALYSIS OF FUEL/CLAD MECHANICAL INTERACTION IN NON-STEADY REACTOR OPERATION VIA FUEL ROD MODELLING CODE IAMBUS. Steinmetz, B.; Gehr, H.L.; Toebe, H. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). pp C4/12 of Structural mechanics in reactor technology. Vol. C. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

BURNUP;FUEL RODS;FUEL-CLADDING INTERACTIONS; I CODES;LMFBR TYPE REACTORS;MECHANICAL PROPERTIES;SIMULATION;STRAINS;STRESSES;WATER COOLED REACTORS

- 5096 SIMULATION STUDY OF THE SECONDARY RUPTURE OF SG TUBING FOR LMFBR. Tsuyuki, T. (Ishikawajima-Harima Heavy Industries Co. Ltd., Yokohama (Japan). Yokohama No.2 Works). pp E5/7 1-10 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

It is known that defects in a heat transfer tube of an LMFBR SG (steam generator) can be a cause of water leakage, which, eroding the adjacent tubes away (wastage), may lead to a large-scale water leakage (water injection) and water-sodium reaction. This paper, which is a part of a large research program for safety design of SG, intends to present and discuss the results of a simulation study made to determine the water injection rate and the tube fracture aspect as a function of the wastage shape.

- 5097 ANALYSIS OF BOWED REACTOR CORES USING THE FIAT-PROGRAM. Urban, K. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch

Gladbach (Germany, F.R.)). pp D2/6 1-11 of Structural mechanics in reactor technology. Vol. D. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The core elements of a LMFBR are bowed due to neutron flux and temperature gradients. The movement of the elements due to free bowing is restricted by neighbouring elements or constraint rings. Consequently forces are induced in the core. To calculate the forces and bowed shapes of the fuel elements at different power and burn-up situations the program FIAT was developed. FIAT is based on the NUBOW code of ANL. FIAT calculates the force equilibrium configuration of elements in a spoke treating the ducts as beam members. The iteration technique, which is taken from NUBOW, is used in FIAT to calculate the gap distribution between the pads of the elements in the considered spoke. Knowing the gap distribution a direct solution of the unknown mechanical displacements and forces in the load planes can be given. As the gap distribution hardly changes during the burn-up history, the use of the iteration part is not required very often. The computer running times of the combined method are about 10 times less than the strict iteration technique as used in NUBOW. The computer running time reduction is improving with increasing numbers of load planes between the elements. Other improvements of FIAT over NUBOW are bowing of elements due to neutron induced swelling and creep effects, and the growth of duct width due to pressure drop and creep (bulging).

- 5098 EFFECT OF LARGE SODIUM-WATER REACTIONS ON LMFBR STEAM GENERATORS: INITIAL RESULTS FROM SWAAM-I. Valentin, R.A. (Argonne National Lab., IL (USA)). pp E5/1 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

FAILURES;LMFBR TYPE REACTORS;MOLTEN METAL-WATER REACTIONS;S CODES;SECONDARY COOLANT CIRCUITS;SODIUM;STEAM GENERATORS;STRESS ANALYSIS;THERMAL ANALYSIS;TRANSIENTS;TUBES

- 5099 EXPERIMENTAL AND THEORETICAL EVALUATION OF DEFECT GROWTH OF FUEL PIN CLADDING BY CHEMICAL FUEL-SODIUM INTERACTION. Weimar, P. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). pp C4/6 1-12 of Structural mechanics in reactor technology. Vol. C. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In the system fuel-sodium-oxygen a reaction product (Na_3MO_x), where $M = \text{U} + \text{Pu}$, is formed after attaining an equilibrium oxygen concentration in the sodium. The reaction product is characterized by a low density (e.g. 5.5 g/cm³) and a low thermal conductivity (about 1/10 of that of the fuel). The melting point is about 1420°C. The reaction was predicted previously by thermodynamic assessment for UO_2 and for mixed oxide. First estimates were made for O-diffusion through the fuel to the defect after failing below the equilibrium O concentration in sodium. Because of the small diffusion coefficient, the calculated diffusion time was large and the swelling rate of the defected pin was

relatively low. There are now available some results of German irradiation experiments with defected pins in sodium. The measured swelling rates are much higher than the former predicted values. With two irradiation experiments of the German Fast Breeder Project SILOE S2 and MOL 7B it is now possible to discuss the effects of chemical fuel sodium reactions. The time dependence upon the swelling reaction was studied with a single pin in the French SILOE reactor and the long term behaviour of a failed 18 pin bundle in a sodium loop in the Belgian BR2 reactor was evaluated. The S2 pin showed a diameter increase of 25% after 50 d of irradiation. The respective values of the bundle after running 180 d beyond failure were some 30%. An evaluation of the pin defects referring to their orientation showed significant blockade in the colder channels of the bundle.

- 5100 2-D THERMOELASTIC ANALYSIS OF LMFBR FUEL ROD CLADDINGS. Wolf, L.; Wong, C.N.; Yeung, M.K. (Massachusetts Inst. of Tech., Cambridge (USA). Dept. of Nuclear Engineering). pp C4/4 1-16 of Structural mechanics in reactor technology. Vol. C. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The primary purpose of this study is twofold. Firstly, unique 2-D and 3-D temperature fields for bare finite rod bundles are presented which result from detailed distributed-parameter analyses using the slug flow model. Secondly, these results together with azimuthally varying inside and outside pressure fields are used to derive thermoelastic stress and displacement fields. The 2-D temperature fields are obtained by an iterative multicell, multiregion technique. Multicell calculations are especially important for studying the pronounced asymmetry effects induced by the coolant cells which are bounded by the bundle wall. The solution takes into account an arbitrary flow split between wall and internal subchannels as well as different heat source densities in the pins. Results indicate that the adiabatic single cell calculation is not conservative under all circumstances as has been previously assumed. Integral mixing transport parameters which account for local temperature variations are quite different from those which are built into the subchannel codes and should be preferred in the future. The pin model used in these studies incorporates the following features: tilted power distribution as experienced across blanket rods; eccentric pellet position as the most probable pellet location at BOL, cracked pellet under higher burnup conditions.

- 5101 MECHANICAL RESPONSE OF FFTF REFERENCE AND P1 CLADDING TUBES UNDER TRANSIENT HEATING. Youngdahl, C.A.; Pierce, R.D.; Lepacek, B.E. (Argonne National Lab., IL (USA)); Arman, T. (Notre Dame Univ., IN (USA)). pp C4/11 of Structural mechanics in reactor technology. Vol. C. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
- From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
- FAILURES;FUEL CANS;MECHANICAL PROPERTIES; PLASTICITY;FFTF REACTOR;STAINLESS STEEL-316; STRAINS;STRAINS;TUBES;VERY HIGH TEMPERATURE

- 5102 INFLUENCE OF SODIUM TEMPERATURE, GAMMA HEATING AND WALL THICKNESS ON DESIGN STRENGTH

OF SLSF CONTAINMENT VESSELS. Thompson, D.H.; Tessier, J.H.; Wang, P.Y.; Lennox, D.H. (Argonne National Lab., IL (USA)). pp G9/7 of Structural mechanics in reactor technology. Vol. G. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

CONTAINMENT SHELLS;INCONEL ALLOYS;LMFBR TYPE REACTORS;RADIATION HEATING;SODIUM; SPECIFICATIONS;STAINLESS STEEL-316;TEMPERATURE DEPENDENCE;THICKNESS

- 5103 BEHAVIOR OF ANNEALED TYPE 316 STAINLESS STEEL UNDER MONOTONIC AND CYCLIC BIAXIAL LOADING AT ROOM TEMPERATURE. Ellis, J.R.; Robinson, D.N.; Pugh, C.E. (Oak Ridge National Lab., TN (USA)). pp L8/4 of Structural mechanics in reactor technology. Vol. L. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

ANNEALING;ELASTICITY;LMFBR TYPE REACTORS; MEDIUM TEMPERATURE;PLASTICITY;REACTOR COMPONENTS;STAINLESS STEEL-316;STRAIN HARDENING; STRESS ANALYSIS;STRESSES

- 5104 SOME TRENDS IN CONSTITUTIVE EQUATION MODEL DEVELOPMENT FOR HIGH-TEMPERATURE BEHAVIOR OF FAST-REACTOR STRUCTURAL ALLOYS. Pugh, C.E.; Robinson, D.N. (Oak Ridge National Lab., TN (USA)). pp M6/5 of Structural mechanics in reactor technology. Vol. M. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

CHROMIUM-MOLYBDENUM STEELS;CREEP;DEFORMATION; EQUATIONS OF STATE;LMFBR TYPE REACTORS; MATHEMATICAL MODELS;REACTOR COMPONENTS; STAINLESS STEEL-304;STAINLESS STEEL-316;STRAIN HARDENING

- 5105 DYNAMIC RESPONSE OF SINGLE HEXAGONAL LMFBR CORE SUBASSEMBLY WRAPPERS. Ash, J.E.; Marciniak, T.J. (Argonne National Lab., IL (USA)). pp E3/5 1-11 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

To analyze the dynamic structural response of the LMFBR core subassembly hexagonal wrappers to postulated local energy releases and the sensitivity of the response to variations in both the pressure loading and the material properties of the stainless steel, a finite-element computer code STRAW has been developed. Supplementing the analysis, out-of-pile experiments have been performed on single hexcan sections subjected to internal pressure pulse loadings. The tests have been designed with the purpose of validating, and where necessary, extending the scope of the code. The dynamic pressure loading was generated by an explosive source. The source was designed to produce conditions simulating those of a hypothetical core accident. An example chosen was the hypothesized pressure pulse created by a fuel-coolant interaction resulting from the failure of a misloaded, overenriched fuel pin. A series of experiments was performed to study the effects of variations in material properties.

- 5106 RESPONSE OF SIMPLE, SCALE MODEL REACTOR VESSELS TO A SIMULATED HCDA LOADING. Cagliostro, D.J.; Romander, C.M.; Florence, A.L. (Stanford Research Inst., Menlo Park, CA (USA). Poulter Labs.). pp E2/4 1-12 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

To help evaluate the safety of LMFBRs, computer codes have been developed for predicting the structural response of LMFBRs to HCDAs. Validation of these codes is best done by performing experiments with simple scale models of a reactor. The work described here was performed to verify modeling techniques used in two-dimensional axisymmetric fluid-structure-interaction codes. A well-defined energy source with precision tolerances, and simple boundary conditions and the loading, strain, and deformations were measured. The experiments were performed in thick-walled and thin-walled vessels that are approximately 1/30-scale of the Clinch River Breeder Reactor (CRBR). The results of these experiments are presented in a form suitable for verifying two-dimensional fluid-structure codes designed to predict the response of reactor vessels to HCDA loads.

- 5107 VARIABILITY OF CLADDING DEFORMATION OF LMFBR FUEL ELEMENTS. Hofman, G.L. (Argonne National Lab., IL (USA)). pp C4/5 of Structural mechanics in reactor technology. Vol. C. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
- From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
- CARBIDES; CREEP; EBR-2 REACTOR; FUEL CANS; LMFBR TYPE REACTORS; PHASE TRANSFORMATIONS; PHYSICAL RADIATION EFFECTS; PRECIPITATION; RAPSIDIE REACTOR; STAINLESS STEELS; DEFORMATION; SWELLING

- 5108 ANALYTICAL STUDIES OF A FINITE SIX-CHANNEL BLOCKAGE IN A LMFBR SUBASSEMBLY. Miles, K.J.; Ragland, W.A.; Marr, W.W. (Argonne National Lab., IL (USA)). pp C4/10 of Structural mechanics in reactor technology. Vol. C. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
- From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
- BOUNDARY CONDITIONS; FUEL ASSEMBLIES; FUEL CANS; HEAT TRANSFER; LMFBR TYPE REACTORS; LOSS OF FLOW; REACTOR CHANNELS; MELTDOWN; S CODES; T CODES; THREE-DIMENSIONAL CALCULATIONS

- 5109 EFFECT OF HIGH EXTERNAL THERMAL LOADING ON LMFBR SUBASSEMBLY INTEGRITY. Smith, J.L.; Helenberg, H.W.; Kilsdoun, D.J.; Pedersen, D.R.; Pierce, R.D.; Wilson, R.E. (Argonne National Lab., IL (USA)); Ariman, T. (Notre Dame Univ., IN (USA)). pp E3/9 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
- From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
- COMPARATIVE EVALUATIONS; COMPUTER CODES; DUCTS; FUEL ASSEMBLIES; LMFBR TYPE REACTORS; LOSS OF FLOW; MELTDOWN; REACTOR COMPONENTS; SIMULATION

- 5110 ELASTIC AND ELASTIC-PLASTIC BUCKLING OF VESSEL HEADS. COMPUTATION BY THE CEASENT

SYSTEM. Canton, B. (Institut Universitaire de Technologie, F-00000 Cachan, France); Hoffmann, A.; Roche, R.L. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. des Etudes Mecaniques et Thermiques); Troclet, C. (Ecole Nationale Supérieure de Mecanique et d'Aerotechnique (ENSMA), 86 - Poitiers (France)). pp G7/4 1-11 of Structural mechanics in reactor technology. Vol. G. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In Liquid Metal Fast Breeder Reactors of integrated type, the sodium pressure is low, but the dimensions are large. The thickness-diameter ratios are rather smaller than in other type of nuclear pressure vessels. Therefore buckling must be considered as a potential mode of failure. For instance, the bottom head of the main vessel is loaded by internal hydrostatic pressure and dead weight, and there is a need for buckling analysis. It must be pointed out that in practical cases, buckling occurs after plastic deformation. Due to the effect of initial imperfections elastic buckling is rather difficult to analyse. Moreover, elastic-plastic buckling theory is not a very clear theory and consideration must be given to finite deflexions (and perhaps to initial stresses). So the choice of a suitable mathematical model is a matter of good engineering judgment. In the CEASENT system of structural analysis by finite element method, a special program has been developed for that purpose. The model chosen (MOTAN) is near a tangent modulus model, because it seems a better approximation of the real behaviour of mechanical structures. The buckling load of shells of any shape can be predicted. The method is now operational but a validation program is still in progress.

- 5111 THERMAL ELASTO-PLASTIC ANALYSIS IN REACTOR VESSEL NOZZLES. Cesari, F.; Gradari, P.L. (Comitato Nazionale per l'Energia Nucleare, Bologna (Italy). Programma Reattori Veloci); Curioni, S. (Bologna Univ. (Italy)). pp G9/8 1-8 of Structural mechanics in reactor technology. Vol. G. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
- From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
- In a sodium cooled nuclear fast reactor the thermal analysis has a particular importance as a consequence of the rapid transients and the high sodium conductivity. The aim of this work is to estimate the value of transient thermal stresses in elastic and plastic fields. In the first part, the loading conditions for different steady and transient operations are examined. It is not easy to evaluate theoretically the temperature distribution in the vessel wall as a consequence of different core transients. For this reason, a number of tests have been placed, which simulate in a mock-up the temperature changes in the upper part of the vessel for different thermal transients considered most dangerous: scram to full power, electric supply fault in a pump with reactor scram, pump seizure with reactor scram. In the second part of the work the thermo-elastic structural response is considered using the FEM analysis. The elastic approach has had two functions: the first is to find a simplified structure in two dimensions that has the same behavior as the 3-D nozzle; the second is to follow the thermal transient to look for the time of maximum stress.

- 5112 ELASTOPLASTIC AND NONSYMMETRIC BUCKLING ANALYSIS BY FINITE ELEMENT OF SUPER-PHENIX PRIMARY VESSEL HEAD. le Guern, M.; de Rouvray, A.L. (Engineering System International S.A., 20, Rue Saarinen, G-94150 Rungis-silic, France); Gerij, M. (Compagnie d'Ingenierie pour les Reacteurs au Sodium (CIRNA), Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)); Esztergar, E.P. (Consultant, 442 Westbourne Street, La Jolla, California 92037, U.S.A.). pp G7/5 of Structural mechanics in reactor technology. Vol. G. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

ASYMMETRY; DEFORMATION; EARTHQUAKES; ELASTICITY; FINITE ELEMENT METHOD; LMFBR TYPE REACTORS; P CODES; PLASTICITY; REACTOR VESSELS; THREE-DIMENSIONAL CALCULATIONS; SUPER PHENIX REACTOR

- 5113 THERMAL RATCHETTING IN PRESSURE VESSELS. Tonarelli, L.C.; Ticozzi, C. (Nuclear Component Department, Breda Termomeccanica S.p.A., Viale Sarca, 336, I-20100 Milano, Italy). pp G9/6 1-12 of Structural mechanics in reactor technology. Vol. G. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

During its lifetime a nuclear power plant undergoes cyclic thermal loadings, i.e. time variations of temperature spatial distribution; these thermal cycles combined with mechanical loads (pressure), can produce dimensional changes for creep, plastic cycling and/or ratchetting. This problem is in special way important for a Liquid Metal Fast Breeder Reactor (LMFBR) because temperatures in it are high (about 500°C) and change either in radial direction or in axial direction. While the plastic behaviour of a structure subjected to mechanical loads and/or mechanical cycle loads can be well studied by Finite Element analyses and several different computer programs, the thermal plastic behaviour was studied for simplified conditions only. Actually, the Bree, Miller, Burgreen model considers a thin cylindrical shell with a radial thermal gradient and in uniaxial stress state while Kalnins, Updike extend this model to a biaxial stress state. To obtain numerical results of the thermo-plastic behaviour of cylindrical shells, a mathematical model was developed and the special purpose computer program PLAST was implemented.

- 5114 ACOUSTIC WAVE MEASUREMENTS IN REACTOR-GRADE CONCRETES. Sutherland, H.J.; Kent, L.A. (Sandia Labs., Albuquerque, NM (USA)). pp E6/5 1-10 of Structural mechanics in reactor technology. Vol. E. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

In this paper, five reactor grade concretes were investigated during their initial cure period to gather information which is a suitable data base for interpreting the results of confirmatory reactor safety research programs. In particular, the concretes studied were the concrete proposed for the Clinch River Breeder Reactor (CRBR), which is a limestone mix with flyash, and two other limestone aggregate mixes and two basaltic-aggregate mixes. The curing process in these concretes

was followed with strength tests and with acoustic-wave velocity measurements. The choice of the latter technique was based on the successful programs of several investigators who used acoustic waves to follow the curing process in concrete and this information was also required for instrumentation techniques that are proposed for accident simulation studies.

- 5115 PROCESS INSTRUMENTS FOR SODIUM COOLED FAST REACTORS. Hunjan, A.S.; Rahman, O.A. (Reactor Research Centre, Kalpakkam (India). Fast Reactor Group). pp 11-16 of Nuclear reactor instrumentation. Vol. 2, Sect. 1. Bombay; Department of Atomic Energy (1977).

From Symposium on nuclear reactor instrumentation; Bombay, India (20 Dec 1976).

Liquid sodium is chosen as the best suited coolant for fast breeder reactors because of its low vapour pressure and good heat transfer properties. But liquid sodium reacts violently with air and water and is corrosive. It also acquires considerable radioactivity while passing through the reactor at temperatures as high as 600deg C. Hence, liquid sodium instruments have to be designed with great constraints in view; such instruments with sensor-in-pocket arrangement, for measurement of flow, level, pressure and temperature are discussed.

- 5116 INSTABILITY ANALYSIS OF ELBOWS IN THE PLASTIC RANGE. Sobel, L.H.; Newman, S.Z. (Westinghouse Electric Corp., Madison, PA (USA). Advanced Reactors Div.). pp L3/2 1-13 of Structural mechanics in reactor technology. Vol. L. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The elbow components of a LMFBR piping system have a high degree of flexibility in comparison with that of the straight components. This flexibility may result in severe strain concentrations and instability of the elbow. This work presents some basic considerations underlying the instability in the plastic range of typical LMFBR elbows (radius-to-thickness ratio $r/t=20-50$), and the impact of these considerations on the practical aspects of solution by computer methods. The elbows are made from a strain hardening material and they are loaded by in-plane closing or opening bending moments. A simplified method of computer analysis is used and the particular type of instability considered is that of geometric buckling into a mode that is characterized by significant ovalization of the cross section and by a relatively gradual variation in shape along the length of the elbow. Emphasis is placed on the effects of including geometric nonlinearities (large displacement terms) in the inelastic (time-independent) analysis. Numerical results are presented for a typical LMFBR elbow.

- 5117 CREEP BUCKLING OF SHELLS. Stone, C.M.; Nickell, R.E. (Sandia Labs., Albuquerque, NM (USA)). pp L7/8 1-12 of Structural mechanics in reactor technology. Vol. L. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Thin-walled structural members that are subjected to sustained loads while at elevated temperature may exhibit a time-dependent structural instability, referred to as creep

buckling. Because of the characteristics of LMFBR primary piping components (thin-walled, low pressure, high temperature), the designer must guard against creep buckling as a potential failure mode for certain critical regions, such as elbows, where structural flexibility and inelastic response may combine to concentrate deformation and cause instability. The ASME Boiler and Pressure Vessel Code through its elevated temperature Code Case 1592 (Section III, Division 1) provides design rules for Class 1 components aimed at preventing creep buckling during the design life. A similar set of rules is being developed for Class 2 and 3 components at this time. The creep buckling design rules were altered in order to accommodate load and strain factors of safety that were thought to be roughly equivalent to the factor of ten on service life. The specific concern is for the case of load-controlled time-dependent buckling where the load factor is 1.5 for normal, upset, and emergency operating conditions, and 1.25 for faulted operating conditions. The current study was conducted in an effort to determine the conservatism or lack of conservatism in current ASME design rules concerning time-dependent (creep) buckling. The two rules are examined from the standpoint of the materials behavior - specifically the secondary creep strain rate exponent.

- 5118 CREEP-FATIGUE DESIGN CRITERIA FOR APPLICATION TO A 2 1/4 CR-1 MO LMFBR STEAM GENERATOR. Campbell, R.D. (Engineering Decision Analysis Company, Inc., Irvine, CA, USA); Veljovich, W. (Rockwell International Corp., Canoga Park, CA (USA)). pp L7/2 1-11 of Structural mechanics in reactor technology. Vol. L. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
The existing ASME Boiler and Pressure Vessel Code, Code Case 1592-7, does not contain a complete set of rules for design of Class 1 elevated temperature components constructed of 2 1/4 Cr-1 Mo alloy steel. General criteria for evaluation of the components for deformation limits and creep-fatigue interaction are contained, but the required fatigue curves, interaction diagram, and K value are not provided. The Code Case allows the use of an alternate criterion by the Manufacturer, provided that such criterion is incorporated into the Owner's Design Specifications. Consequently, a study was undertaken to evaluate alternate criteria for creep-fatigue evaluation of LMFBR steam generators constructed of 2 1/4 Cr-1 Mo alloy steel.

- 5119 PROPERTY CHANGES IN FAST-REACTOR CORE-SUPPORT COMPONENTS DUE TO LOW DOSE RATE, LONG-TIME IRRADIATION. Johnson, D.L.; Hofman, G.L. (Argonne National Lab., IL (USA)). pp L8/7 1-10 of Structural mechanics in reactor technology. Vol. L. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
To ensure the integrity of LMFBR core-support structures, it is necessary to accurately predict the magnitudes and directions of the mechanical and physical property changes known to occur in structural materials during irradiation. Since the neutron-energy spectrum is relatively soft and the total neutron flux relatively low in the vicinity of the support structures, little

directly applicable data are obtainable. An analytical technique, such as the damage-function analysis, must be used to extrapolate from harder spectra data. This technique has been used to analyze the grid-support structure in the Experimental Breeder Reactor-II.

- 5120 FINITE ELEMENT ELASTIC-PLASTIC ANALYSIS OF LMFBR COMPONENTS. Levy, A.; Pifko, A.; Armen, H. Jr. (Grumman Aircraft Engineering Corp., Bethpage, NY (USA). Research Dept.). pp L4/4 of Structural mechanics in reactor technology. Vol. L. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
CREEP;ELASTICITY;FINITE ELEMENT METHOD;LMFBR TYPE REACTORS;PLASTICITY;REACTOR COMPONENTS; SHELLS;STRAINS;STRESS ANALYSIS;THERMAL STRESSES

- 5121 INELASTIC RESPONSE OF METALLIC IMPACT LIMITERS. Biffle, J.H.; Sagartz, M.J.; Weingarten, L.I. (Sandia Labs., Albuquerque, NM (USA)). pp J6/10 1-12 of Structural mechanics in reactor technology. Vol. J, Pt. b. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
Energy mitigators on some light water reactor (LWR) spent fuel shipping casks and conceptual designs for liquid metal fast breeder reactor (LMFBR) spent fuel shipping casks rely on the inelastic behaviour of metallic plate-like structures. A program was undertaken to investigate the response of mitigators to hypothetical shipping accidents with parallel experimental and theoretical studies. The theoretical studies consist of performing large deformation three dimensional transient dynamic analysis on several metal fin designs and correlating the results with the experimental results. The newly developed computer program WULFF was used. Testing of scale model fin specimens was performed at SLA's drop table facility. The test specimens are mounted on a force transducer and this assembly is bolted to the stationary bottom table. Extra weights are added to the drop table so that the total energy at impact is sufficient to cause large plastic deformation in the specimens. Accelerometers mounted on the drop table and the data from the force transducer are used to compute the energy absorbed during the test. Also framing camera coverage was used to observe the specimen deformation as it develops. The results of the theoretical analysis and experimental program can be used by designers responsible for impact limiters on spent fuel shipping casks. Without data to design impact limiters the integrity of the cask under accident conditions can be questionable. 9

- 5122 CONTAINMENT OF THE CREYS-MALVILLE PLANT. Noel, H.; Fourest, B.; Journet, J.; Lallient, J. (Compagnie d'Ingenierie pour les Reacteurs au Sodium (CIRNA), Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)). pp J1/4 of Structural mechanics in reactor technology. Vol. J, Pt. a. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).
From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).
BUILDINGS;CONTAINMENT SYSTEMS;FUEL CANS; REACTOR SAFETY;REACTOR VESSELS;SUPER PHENIX

REACTOR

- 5123 SOME COMMENTS ON THERMOMECHANICAL CONSTITUTIVE EQUATIONS FOR INELASTIC ANALYSIS OF LMFBR COMPONENTS. Haupt, P. (Bundesanstalt fuer Materialpruefung, Berlin (Germany, F.R.)). pp L1/6 1-13 of Structural mechanics in reactor technology. Vol. L. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

The safety analysis of the primary and secondary circuits of sodium-cooled fast breeder reactors (LMFBR) requires systematic methods to predict the inelastic behaviours of metal structures under complex thermomechanical loading conditions. To provide those methods, constitutive equations have been recommended for interim use. Essentially, the recommended constitutive theories are based on the additive decomposition of the total strain into elastic, plastic and viscoelastic parts. These parts are calculated independently according to Hooke's law, classical plasticity and particular creep laws. This paper is concerned with the physical evidence and consistency of these constitutive equations. In particular, shortcomings of the recommended constitutive theories may result from the bilinear approximation of the stress-strain-curve; the kinematic linear hardening rule and the very special kind of the proposed creep laws. Moreover, the elastic-plastic and creep equations are formulated independently: Except for hardening due to prior creep strains, possible interactions between viscoelastic and plastic material behaviors are neglected. Finally, the temperature appears as a given parameter, that means: The recommended constitutive assumptions do not allow for any coupling effects between the deformation and temperature field. A consistent set of constitutive equations should be based on a correct evaluation of the second law of thermodynamics.

- 5124 PLASTIC CREEP BEHAVIOUR ANALYSIS OF AN ELBOW PIPE. Bourrier, P. (Department Genie Industriel et Nucleaire, SOCOPEC, Paris, France); Hoffmann, A. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)). pp L6/2 1-12 of Structural mechanics in reactor technology. Vol. L. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

High temperature components are liable to strong creep effects, in LMFBR for example. This phenomenon is analyzed here for an elbow pipe. Comparisons are performed between experiments and computations using CEASEMT codes, particularly piping-system analysis code TEDEL.

- 5125 ANALYSIS OF PIPING SYSTEMS FOR CREEP. Boyle, J.T.; Spence, J. (Strathclyde Univ., Glasgow (UK). Dept. of Mechanics of Materials). pp L2/7 1-14 of Structural mechanics in reactor technology. Vol. L. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Until recently piping system design was justified in terms of purely elastic analysis principles. In evaluating the safety and reliability of high temperature systems, in

particular those associated with LMFBR development, inelastic behaviour must be admitted and the possibility of the accumulation of strain in the highly stressed locations accounted for. The purpose of this paper is to provide a survey of the current situation and to attempt to explain and clarify a number of misconceptions which have arisen in connection with approximative techniques. The paper analyses the motivation for a simplified approach to the initial stages of elevated temperature piping design, examines the implications for such an approach and provides a possible solution to the problems which arise.

- 5126 EVALUATION OF INELASTIC ANALYSIS METHODS. Corum, J.M. (Oak Ridge National Lab., TN (USA)). pp L4/1 1-12 of Structural mechanics in reactor technology. Vol. L. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

Interim constitutive equations have been developed for describing the inelastic behaviors of types 304 and 316 stainless steel and 2 1/4 Cr-1 Mo steel. Uniaxial and multiaxial exploratory tests were used to identify the essential inelastic behavioral features of significance, and these features, in turn, provided the basis for choosing the interim constitutive equations. Typical exploratory test results are presented to support the basic constitutive equation framework and to illustrate how the ad hoc rules were developed, evaluated, and improved. To assess the validity of the constitutive equations and analysis methods for high-temperature design, benchmark structural tests were conducted under time-varying loadings at elevated temperatures, and the results were compared to inelastic analysis predictions. Representative test results for beam and plate structures and for a pipe subjected to repeated thermal shocks are presented and compared to predictions. Although these comparisons are generally favorable, they did identify some specific analysis shortcomings. These are discussed, and procedures for improving the predictions are presented. Finally, the effects that normal material properties variations can have on analysis predictions are discussed and shown to be significant.

- 5127 PROBABILISTIC ASSESSMENT IN STRUCTURAL ANALYSIS OF MAIN COOLANT SYSTEMS FOR LMFBR UNDER ANTICIPATED THERMAL TRANSIENTS. Amesz, J.; Volta, G. (Commission of the European Communities, Ispra (Italy). Joint Research Centre); Cesari, F.G.; Righini, R. (Comitato Nazionale per l'Energia Nucleare, Bologna (Italy)). pp M9/8 of Structural mechanics in reactor technology. Vol. M. Jaeger, T.A.; Boley, B.A. (eds.). Amsterdam, Netherlands; North-Holland (1977).

From 4. international conference on structural mechanics in reactor technology; San Francisco, CA, USA (15 Aug 1977).

LMFBR TYPE REACTORS; PRIMARY COOLANT CIRCUITS; PROBABILITY; REACTOR SAFETY; REACTOR SHUTDOWN; SECONDARY COOLANT CIRCUITS; THERMAL STRESSES; TRANSIENTS; STRESS ANALYSIS

- 5128 POWER SUPPLY FOR CONTROL AND INSTRUMENTATION IN FAST BREEDER TEST REACTOR. Raghavan, K.; Shanmugam, T.K. (Reactor Research Centre, Kalpakkam (India)). pp 1-5 of Nuclear reactor instrumentation. Vol. 2, Sect. 2. Bombay, India; Department of Atomic Energy

(1977).

From Symposium on nuclear reactor instrumentation; Bombay, India (20 Dec 1976).

The design and operation of the four 'no-break' power supplies for control and instrumentation in the Fast Breeder Test Reactor (FBTR), Kalpakkam, are described. Interruptions in the power supplies are eliminated by redundancy and battery back-up source while voltage dips and transients are taken care by automatic regulation system. The four power supplies are : (1) 24 V D.C. exclusively for neutronic and safety circuits, (2) 48 V D.C. for control logic indication lamps and solenoid valves, (3) 220 V D.C. for switchgear control, control room emergency lighting and D.C. flushing oil pump for the turbine and (4) 220 V A.C. single-phase 50 HZ for computers and electronics of control and instrumentation. Stationary lead-acid batteries (lead antimony type) in floating mode operation with rectifier/charger are used for emergency back-up. All these power supplies are fed by 415 V, 3-phase, 50 HZ emergency supply buses which are provided with diesel generator back-up. Static energy conversion system (in preference to mechanical rotation system) is used for A.C. to D.C. and also for A.C. to A.C. conversion.

- 5129 DEVELOPMENT OF LEAK-PROOF CABLES FOR SIMPLIFYING CABLE PENETRATIONS ACROSS REACTOR CONTAINMENT BUILDING. Sangodkar, D.B.; Prabhakar, R.; Sivanandam, M. (Reactor Research Centre, Kalpakkam (India)). pp 24-28 of Nuclear reactor instrumentation. Vol. 2, Sect. 2. Bombay, India; Department of Atomic Energy (1977).

From Symposium on nuclear reactor instrumentation; Bombay, India (20 Dec 1976).

All penetrations including cable penetrations of the containment building must be leak-tight so that leakage rate for released radioactive material inside the containment building is kept to a minimum. Special leak-proof cables utilising elastic insulating compounds to seal the intrastand and interconductor space have been developed for the cable penetration assemblies for use in the Fast Breeder Test Reactor, Kalpakkam. More than 50 samples of these cables were tested for leak-tightness and the results are presented. Overall leakage rate for most of the samples of 3 metre length is found to be 0.02%. This rate will be lower by a factor of 10 for an in-field length of 50 meters, because an inverse relationship was observed to exist between cable length and leakage. Leakage is also found to be independent of electrical loading.

- 5130 NUCLEAR FUEL ASSEMBLY. Betten, P.R. (to Combustion Engineering, Inc., New York (USA)). French Patent 2,337,406/A/. 31 Dec 1976. 14p. (In French).

Under the invention the fuel assembly is particularly suitable for liquid metal cooled fast neutron breeder reactors. Hence, according to the invention a fuel assembly cladding includes inward corrugations with respect to the remainder of the cladding according to a recurring pattern determined by the pitch of the metal wire helically wound round the fuel rods of the assembly. The parts of the cladding pressed inwards correspond to the areas in which the wire encircling the peripheral fuel rods is generally located apart from the cladding, thereby reducing the play between the cladding and the peripheral fuel rods situated in these areas. The reduction in the play in turn improves the coolant flow in the internal secondary channels of the fuel assembly to the detriment of the flow in the peripheral

secondary channels and thereby establishes a better coolant fluid temperature profile.

- 5131 SAFETY DEVICE IN A NUCLEAR REACTOR FUEL ROD WITH A CLOSED FISSION GAS PLENUM. Gast, K. (to Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). German (FRG) Patent 2,023,250/C/. 23 Dec 1976. 2p. (In German).

The invention deals with the design of the fission gas plenum of fuel rods for nuclear reactors cooled with liquid metal. The fission gas plenum is subdivided by honeycombed partition walls - preferably of dry powdered metal - being fully permeable for fission gases. In this way a slow, uniform pressure build-up is guaranteed. If the can of the fuel rod fails in this area, the stored amount of fission gas is released only delayed. The invention increases the safety of the fuel elements equipped with the device.

- 5132 FUEL ROD FOR LIQUID METAL-COOLED NUCLEAR REACTORS. Vinz, P. (to Messerschmitt-Boelkow-Blohm G.m.b.H., Muenchen (Germany, F.R.)). German (FRG) Patent 2,449,597/B/. 16 Dec 1976. 5p. (In German).

In fuel rods for nuclear reactors with liquid-metal cooling (sodium), with stainless steel tubes with a nitrated surface as canning, superheating or boiling delay should be avoided. The inner wall of the can is provided along its total length with a helical fin of stainless steel wire (diameter 0.05 to 0.5 mm) to be wetted by hot sodium. This fin is mounted under prestressing and has a distance in winding of 1/10 of the wire diameter.

- 5133 COOLING DEVICE FOR NUCLEAR POWER PLANT. Kawakami, H. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-72,989/A/. 13 Dec 1976. 4p. (In Japanese).

A method to facilitate the calibration for flowmeters in a reactor cooling device after installation and during operation of a power plant is described. In a reactor cooling system for sodium cooled type fast reactor, an injection device for charging an activated sodium flow quantity measurement substance and a plurality of detectors for the γ -radiation emitted from the activated sodium injected are provided in optional portions in the secondary coolant system. The delays in time for the radiation detection of the sodium charged from the injection device are detected to determine sodium flow rate thereby calibrating the flowmeter in the secondary coolant circuit.

- 5134 PIPE CONNECTOR. Sullivan, T.E.; Pardini, J.A. (to Energy Research and Development Administration). US Patent Application 749,206. 9 Dec 1976. 6p.

A safety test facility for testing sodium-cooled nuclear reactor components includes a reactor vessel and a heat exchanger submerged in sodium in the tank. The reactor vessel and heat exchanger are connected by an expansion/deflection pipe coupling comprising a pair of coaxially and slidably engaged tubular elements having radially enlarged opposed end portions of which at least a part is of spherical contour adapted to engage conical sockets in the ends of pipes leading out of the reactor vessel and into the heat exchanger. A spring surrounding the pipe coupling urges the end portions apart and into engagement with the spherical sockets. Since the pipe coupling is submerged in liquid a limited amount of leakage of sodium from the pipe can be tolerated.

- 5135 FUEL ASSEMBLY. Yokoyama, T. (Nippon Atomic Industry Group Co. Ltd., Tokyo). (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-71,784/A/. 8 Dec 1976. 2p. (In Japanese).

A method to reduce useless neutron absorption into a wrapper tube covered a fuel assembly of an LMFBR type reactor by reducing the wall thickness of the wrapper tube is described. The fuel assembly is composed of fuel element clusters and the wrapper tube covering them. The wall thickness of the wrapper tube is changed near the boundary (referred to as transition point) between the core fuel and the blanket fuel. Specifically, the wall thickness of the wrapper tube below the transition point is made uniform determined by the highest inner pressure, which is thinner than the wall thickness above the transition point made uniform determined by the inner pressure at the transition point. By partially decreasing the thickness of the wrapper tube, useless absorption of neutrons into the wrapper tube can be reduced to thereby improve the breeding ratio.

- 5136 DETECTION METHOD FOR THE POSITION OF A SENSOR. Imai, T. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-71,792/A/. 7 Dec 1976. 3p. (In Japanese).

A method for the positioning of a sensor in a guide tube disposed in an LMFBR type reactor at a predetermined position at the top of the guide tube irrespective of the length and the presence or the absence of curvatures in the guide tube is described. A reactor container has a core in which a plurality of nuclear fuel assemblies are arranged and the upper opening of the reactor container is closed with a rotational plug, through which instruments, control drives, a fuel exchanger and the likes are inserted into the container. A sensor for monitoring the flow rate and the temperature of sodium at the exit of the fuel assemblies is inserted through a guide tube and guided to the upper portion of the exit for each of the fuel assemblies. In this invention, electroconductive core wires are exposed at the top of a cable integrated with a sensor and a cable sheath is brazed to the cap. This enables to detect the arrival of the sensor at the top of the guide tube by the changes in the resistance value.

- 5137 SEALING DEVICE. Ugawa, Y. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-71,786/A/. 7 Dec 1976. 5p. (In Japanese).

A method to reduce the damage in an elastomer seal and decrease the driving torque for keeping the sealing performance in a sealing device for cover gas upon rotation and jack up of a rotational shielding plug is described. A shielding plug comprising a stationary plug and a rotational plug is provided to the opening at the upper portion of a reactor vessel to seal the cover gas enclosed in the space above a coolant. A lip type elastomer seal is provided on the stationary plug and a sliding pad member with a concaved contact surface capable of slidably fitting against the lip portion of the elastomer is provided to the rotational plug. As a result, the gap defined between both of the above plugs can completely be sealed closely.

- 5138 EXPANSION JOINT. (to English Electric Co. Ltd., London (UK)). French Patent

2,334,047/A/. 1 Dec 1976. Priority date 2 Dec 1975, United Kingdom of Great Britain and Northern Ireland (UK), 15p. (In French).

The invention concerns the pipe connections used in LMFBR type nuclear reactors. In particular it concerns the expansion joints employed for connecting to the internal components of the reactor (at high temperature) the delivery pipes of the centrifugal pumps for moving the coolant, and sustained by the cooled concrete roof of the containment.

- 5139 EQUIPMENT FOR FBR TYPE REACTOR. Sakai, T. (to Toshiba Corp., Kawasaki, Kanagawa (Japan)). Japanese Patent 1953-68,393/A/. 1 Dec 1976. 3p. (In Japanese).

Purpose: To heat an equipment for use in a fast breeder reactor utilizing a liquid-metal as a coolant prior to the insertion thereof into the reactor thereby preventing the thermal shock from being imparted to the equipment at the time of the insertion thereof and curtailing the period of time of suspending the operation of the reactor. Constitution: When an equipment used in the fast breeder reactor such as fuel exchanger is transported into the reactor, the atmosphere within the cask is preheated with a heater. During this preheating operation these works such as substitution of cover gas within the reactor and lowering of the surface of the sodium solution are advanced in keeping pace therewith, and the preheating operation is stopped at the time when the fuel exchanger is transported into the reactor. By the above described preheating the temperature of the heat exchanger becomes substantially the same as that of sodium within the reactor, and therefore, when the fuel exchanger is dipped in sodium, it does not undergo thermal shock. Furthermore, since the preheating operation can be carried out in the cask, it is possible to curtail the driving rest period, and therefore the driving efficiency of the reactor can be raised.

- 5140 HIGH-TEMPERATURE ACOUSTIC DETECTOR. Miyazawa, T.; Uesugi, N.; Taguchi, J.; Hirano, H. (Tokyo Shibaura Electric Co., Ltd., Kawasaki, Kanagawa, Japan). Toshiba Rebyu; 31: No. 12, 1033-1036 (Dec 1976). (In Japanese).

An acoustic detector usable in high-temperature (700°C) liquid sodium has been developed for abnormality detection in a sodium-cooled fast breeder reactor plant. This detector, incorporating a LiNbO_3 crystal with a high Curie point, is of a piezo system and is so designed as to bear neutron irradiation. The results of its evaluation tests in high-temperature atmosphere, high-temperature liquid sodium and normal-temperature water and its prominent characteristics are as follows: (1) The LiNbO_3 is enclosed hermetically in a welded stainless-steel container; electric signals are transmitted through a coaxial cable for high-temperature use. (2) Its actual detective sensitivity is as designed: -150 dB (0 dB=1 V/ μ bar) in 600°C sodium. (3) Its life has been confirmed to be longer than 12,000 hours at 700°C. (4) Various investigations have been made about the internal construction of the detector container to actualize the above items (2) and (3). (5) In order to prevent performance deterioration due to neutron irradiation, a new kind of crystal has been developed, and now irradiation evaluation tests are being carried on by the cooperation between Toshiba and GfK, West Germany.

- 5141 SAFETY OF HIGH-PARAMETER SODIUM LOOP. Paryzek, R. (Ústav Jaderného Vyzkumu, Rez (Czechoslovakia)). Jad. Energ.; 22: No. 12,

443-447(Dec 1976). (In Czech).

The solution of problems of technical and fire safety is shown on the concrete example of the SMT-1 experimental sodium loop installed at the Institute of Nuclear Research at Rez. The general principles and measures adopted for the loop are described which were derived from a detailed analysis of the possible causes of failures or accidents. These causes include: leakage and cracks arising during production, leakage and cracks caused by the quick temperature changes in operation and by the incorrect heating up of circuits with solidified sodium, equipment failure caused by dilatation forces, by pressure increase above the permissible level and by the corrosive effects of sodium, accidents caused by failures in the measurement and loop control system, and leakage on the main container. The loop is equipped with an alarm system and closed-circuit television.

5142 (SMRE-Transl--7200) METHOD OF COMPUTING THE EFFECT OF GAS VOIDS IN LIQUID METAL COOLING. Malang, S.; Rust, K. Translated from Atomwirtschaft; 579-581(Dec 1976). 8p. Health and Safety Executive, Baynards House, London.

Under certain operating conditions, the coolant entering the reactor tank of loop-type liquid-metal cooled reactors may contain gas bubbles. Constructional measures help to make sure that most of the gas is separated before entering the fuel zone and flows through the radial blanket. The gas bubbles cause the temperature in the fuel rods to rise. The authors present a computer model for the calculation of a two-dimensional, instationary temperature field.

5143 IMPROVEMENT TO STRUCTURES CONTAINING A HOT LIQUID METAL IN BREEDER REACTORS. Chevallier, R.; Sauvage, M. (to Alstom, 75 - Paris (France); CEA, 75 - Paris (France)). French Patent 2,372,494/E/. 25 Nov 1976. 5p. (In French).

The main patent concerns a liquid metal cooled integrated breeder nuclear reactor which includes a main vessel comprising a central internal vessel containing the core and the hot liquid metal. This internal vessel is connected to individual cylindrical containments each possessing a heat exchanger. The surface of the internal vessel, opposite the hot liquid metal flow holes, is given shapes formed by a surface of revolution generated by an arc and the cylindrical containments are secured directly to the internal vessel at this location by joining these two surfaces by an externally facing concave toric surface. In the design forms of the main patent given as examples, the thermal protection of the internal central vessel and the individual cylindrical containments is carried out by two double walls, in the annular space of which cold liquid metal is made to flow. The object of this addendum is a design variant comprising these double walls, in which the thermal protection of the internal vessel and of the cylindrical containment is assured by a quasi static sheet of liquid metal topped by a static layer of a neutral gas in the annular space formed by these walls.

5144 METHOD AND APPARATUS OF STABILIZING METALLIC SODIUM. Nakajima, N.; Tomioka, Y.; Saigusa, M.; Suzuki, K. (to Mitsubishi Atomic Power Industries, Inc., Tokyo (Japan)). Japanese Patent 1978-65,598/A/. 24 Nov 1976. 4p. (In Japanese).

To safely treat a metallic sodium containing

radionuclide exhausted from a liquid-metal cooled fast breeder reactor, a novel method and apparatus for converting the metallic sodium chemically stable sodium hydroxide is described. By alloying metallic sodium with a low melting metal, while the Na-H₂O reaction velocity is being controlled, water decomposition treatment of the metallic sodium is made possible. Since low melting metal remains substantially in a mass state or in a powdery state after Na-H₂O reaction, it becomes possible to recover. Furthermore, a general waste liquid treating system which has heretofore been used is applicable for a heavy metal and radionuclide contained in the waste liquid produced in the case where non-radioactive sodium and sodium containing radionuclide have been treated.

5145 LIQUID METAL COOLED REACTOR. Tanaka, N.; Onogi, T. (to Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Japanese Patent 1978-62,098/A/. 16 Nov 1976. 3p. (In Japanese).

Purpose: To mount a heat-exchanging pipe at the external of a cooling boundary and provide a cooler in the circuit of the heat-exchanging pipe, thus obtaining a small and efficient auxiliary cooling system in a reactor. Constitution: In a normal state, heat generated in the core is removed by a primary coolant from an inlet nozzle of a reactor container, and the heated primary coolant flows into an intermediate heat exchanger from an outlet nozzle of the container through the pipe. Then, the primary coolant is heat-exchanged with a secondary coolant, and the cooled primary coolant is again fed into the container. In the event the cooling system fails, the cooler is actuated to feed a cooled thermal medium to a circuit formed by the heat exchanging pipe, thereby lowering temperature of liquid metal coolant flowing the primary cooling pipe to remove decay heat in the core. On the other hand, a heater maintains a temperature of the liquid metal at a level above a melting point.

5146 (EURFNR--1405) TIME CREEP TESTS ON IRRADIATED SPECIMENS FOR THE MOL 1AL RADIATION PROGRAM. Schirra, M.; Seith, B. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). 12 Nov 1976. Translation of PSB--521 35p. Dep. NTIS, PC A03/MF A01.

The rupture and creep behavior of unirradiated specimen from two 12 percent Cr-steels were tested at 600°C and specimen from a modified AISI 316 LN at 700°C. The influence of the hardening-temperature for one 12 percent Cr-steel was examined. Metallographic examination shows the fracture behavior and changes in the structure.

5147 LOADING METHOD OF CORE CONSTITUTING ELEMENTS. Kasai, S. (Nippon Atomic Industry Group Co. Ltd., Tokyo). (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-60,485/A/. 11 Nov 1976. 3p. (In Japanese).

The purpose is to provide a remote-controlled replacing method for core constituting elements in a liquid-metal cooling fast breeder, wherein particularly, the core constituting elements are prevented from being loaded on the core position other than as designated. The method comprises a first step which determines a position of a suitable neutron shielding body in order to measure a reference level of complete insertion of the core constituting elements, a second step which

inserts a gripper for a fuel exchanger, a third step which decides stroke dimensions of the complete insertion, and a fourth step which discriminates the core constituting elements to begin handling of fuel rods. The method further comprises a fifth step which determines a loading position of fuel rod, and a sixth step which inserts and loads fuel rods into the core. The method still further comprises a seventh step which compares and judges the dimension of loading stroke and the dimension of complete inserting stroke so that when coincided, loading is completed, and when not coincided, loading is not completed and then the cycle of the fourth step is repeated.

- 5148 NUCLEAR REACTOR. Lambert, P. (Alstom, 75 - Paris (France)); Aubert, M. (to CEA, 75 - Paris (France)). French Patent 2,370,344/E/. 3 Nov 1976. 11p. (In French).

This improvement concerns a system for cooling the reactor jacket of a liquid sodium cooled integrated fast reactor. The reactor jacket is located inside a second vessel called the 'main vessel', the reactor jacket being suspended from the upper shield and comprising a first ring and a second ring. The upper edge of the first ring is free and the lower edge is connected and sealed to the reactor vessel which has apertures above the connection level of this first ring. The annular space between the reactor vessel and the first ring is the removal annular space. The annular space between the second ring on the one hand and the reactor jacket and the first ring on the other is the annular supply space.

- 5149 (EURFNR--1404) CONSIDERATIONS FOR THE FUTURE DEVELOPMENT OF SHELL- AND CORE-STRUCTURAL MATERIALS FOR FAST BREEDERS. Anderko, K.; Wassilew, C. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Material- und Festkoerperforschung). Nov 1976. Translation of KFK--2369 33p. Dep. NTIS, PC A03/MF A01.

Work performed under United States--Euratom Fast Reactor Exchange Program.

The report gives (1) a survey on the projected or already progressing development work for the improvement of shell and core structural materials for Fast Breeder Reactors in the Debene countries, and (2) a detailed discussion of one of the property aspects, the creep strength depending on time. After a look at the creep rupture behavior of irradiated materials and at the role played by inter- and intragranular precipitations, and optimization program for the austenitic alloy X10 NiCrMoTiB 15 15 (1.4970) is outlined. The published data on the residual ductility after high neutron doses at short-term and creep loadings are reviewed.

- 5150 DESIGN BASIS OF AND MATERIALS REQUIREMENT FOR SODIUM HEATED STEAM GENERATORS FOR FAST REACTORS. Chetal, S.C.; Sabherwal, M.C. (Reactor Research Centre, Kalpakkam (India)). NHL (Natl. Metall. Lab.) Tech. J. (Jamshedpur, India); 18: No. 4, 73-80 (Nov 1976).

An attempt has been made to categorise the various materials which have been considered for a sodium heated steam generator. Choice of steam parameters, choice of sodium temperature, selection of material, problems of sodium heated generators, R and D efforts for indigenous materials for steam generators etc. have been discussed. It is concluded that stabilised ferritic steel will be a good selection as a sodium heated steam generator

material under Indian conditions.

- 5151 FUNDAMENTAL EXPERIMENT OF POTASSIUM HEAT EXCHANGER USING PRINCIPLE OF HEAT PIPE. Sumida, I.; Kotani, K. (Hitachi Ltd, Ozenji, Kawasaki-shi, Jpn). J. Nucl. Sci. Technol. (Tokyo); 13: No. 1, 648-655 (Nov 1976).

A potassium heat exchanger is described which features separate construction of primary and secondary coolant systems. A small amount of potassium plays a role as an intermediate media of heat transportation. Heat is transferred by evaporation and condensation of potassium on the surfaces of the primary and secondary coolant pipings, respectively. The tests were performed in the temperature range of 200 to 300°C and the maximum heat transfer reaches 1.3 kW (heat transfer rate at the primary heating source: 8.6 W/cm² at 300°C. 8 refs.

- 5152 (JAPFNR--259) STUDY ON SODIUM VOID REACTIVITY WORTH PREDICTIONAL METHOD. Kamei, T.; Yoshida, T.; Murakami, K.; Iijima, S. (Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan); Nippon Atomic Industry Group Co. Ltd., Kawasaki, Kanagawa. Nuclear Research Lab.). Nov 1976. Translation of N--251-76-28 31p. Dep. NTIS (US Sales Only), PC A03/MF A01.

Work performed under United States--Japanese Fast Reactor Exchange Program.

The accurate evaluation of sodium void reactivity of a fast breeder reactor is of great importance in connection with a reactivity addition at high rates, which could cause a large nuclear energy release, in a hypothetical sodium loss accident. However, the complexity of the phenomena has prevented a reliable estimation of the sodium void reactivity worth. This paper presents a detailed analysis of sodium void experiments and assesses the adequacy of the calculational method, and then predicts the sodium void reactivity worth of the Japanese proto-type fast breeder reactor MONJU.

- 5153 MOCK-UP CRITICAL EXPERIMENTS FOR PROTOTYPE FAST BREEDER REACTOR 'MONJU'. Zukeran, A.; Inoue, T.; Suzuki, T.; Kawashima, K. (Power Reactor and Nuclear Fuel Development Corp., Oarai, Ibaraki (Japan). Oarai Engineering Center). Nippon Genshiryoku Gakkaishi; 18: No. 11, 672-684 (Nov 1976). (In Japanese).

The mock-up criticality experiments for Monju are roughly divided into the full mock-up test using the ZEBRA of Winfrith Institute, UKAEA, and the partial mock-up experiment with FCA of JAERI. The former test has been carried out over 18 months from September 1971 as the Japan-UK cooperative research project MOZART. With the FCA, the experiment complementing the MOZART has been carried out, focusing on the nuclear characteristics of Monju which can be simulated with a relatively small core, and the experiment on highly enriched control rods and shielding is being continued now with the FCA 7 core. The experimental data of the MOZART and the ZPPR series in USA were exchanged at the international symposium in Tokyo, thus the prediction and the accuracy evaluation of the nuclear characteristics of Monju became possible, and the highly reliable core design was able to be accomplished. The simulated criticality experiment is necessary for directly grasping the reliability of calculated values in comparison with the experimental values, and also for the experimental prediction of the nuclear characteristics. The outline and the analysis of the simulated

criticality experiment such as reactivity factor, control rod value, reaction rate distribution and sodium void reactivity are described, and the reflection of the results to the design of the core of Monju is explained.

- 5154 IN-CORE NEUTRON INSTRUMENTS. Takeda, T. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1978-56,495/A/. 29 Oct 1976. 3p. (In Japanese).

The purpose is to permit measurement of the neutron flux distribution within a central portion of the core of a fast reactor. A neutron detector is disposed within a cavity formed in a support rod, which is inserted in and secured to a control rod channel in a wrapper tube of a fast reactor. Since the neutron detector is disposed at the center of the core reactor, it is possible to detect the neutron flux in a zone where the void coefficient of sodium is positive, thus permitting accurate measurement of the neutron flux distribution within the core.

- 5155 CORE COOLING SYSTEM FOR REACTOR. Kondo, R.; Amada, T. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1978-54,691/A/. 27 Oct 1976. 5p. (In Japanese).

The purpose is to improve the function of residual heat dissipation from the reactor core in case of emergency by providing a secondary cooling system flow channel, through which fluid having been subjected to heat exchange with the fluid flowing in a primary cooling system flow channel flows, with a core residual heat removal system in parallel with a main cooling system provided with a steam generator. Heat generated in the core during normal reactor operation is transferred from a primary cooling system flow channel to a secondary cooling system flow channel through a main heat exchanger and then transferred through a steam generator to a water-steam system flow channel. In the event if removal of heat from the core by the main cooling system becomes impossible due to such cause as breakage of the duct line of the primary cooling system flow channel or a trouble in a primary cooling system pump, a flow control valve is opened, and steam generator inlet and outlet valves are closed, thus increasing the flow rate in the core residual heat removal system. Thereafter, a blower is started to cause dissipation of the core residual heat from the flow channel of a system for heat dissipation to atmosphere.

- 5156 PIPE PLATE AND ASSEMBLY FOR HEAT EXCHANGER. (to Neratoom N.V., The Hague (Netherlands)). Netherlands Patent 7,611,676/A/. 25 Oct 1976. 6p. (In Dutch).

The pipe assembly, especially for heat exchangers, consists of a pair of horizontal plates connected by a cylindrical tube with a side entry port connected with the space between the plates. The tube has a thicker section and smaller internal diameter at the ends where it holds a smaller outer diameter cylindrical tube. The end joints of the tubes and plates are soldered. The design is relevant to detection of leaks in sodium cooled nuclear power reactors.

- 5157 DOUBLE ARM HANDLING DEVICE FOR NUCLEAR REACTOR. Allain, A.; Buret, J. (to CEA, 75 - Paris (France)). French Patent 2,368,121/A/. 15 Oct 1976. 11p. (In French).

The invention concerns a machine which, inside a liquid sodium cooled reactor, makes it possible to transfer fuel assemblies forming the core, between the positions that these

assemblies fill inside the core to a storage position for these fuel assemblies inside the vessel. These operations are carried out when the reactor plug is in position and when the vessel is filled with liquid sodium, hence they are carried out without being seen. The improved charging system enables the handling time to be cut to a minimum and to diminish the number of positioning operations of the handling system inside the reactor vessel to effect the various charging and discharging operations. It includes a vertically positioned drum of which the top end crosses the plug; a first truck guided in vertical translation by guiding facilities integral with the drum and a second truck guided in vertical translation by other guiding facilities also integral with the drum, each truck being fitted with a horizontal arm at the free end of which is placed a grab hook for holding the fuel assemblies, the guiding facilities being such that the arms remain parallel to each other; gear for moving the trucks vertically and separately; facilities for controlling the opening and closing of the grabs and gear for turning the drum vertically about its centre line.

- 5158 SYSTEM FOR CHARGING AND DISCHARGING THE FUEL OF A NUCLEAR REACTOR. Allain, A.; Buret, J.; Plagnard, A. (Societe Technique pour l'Energie Atomique (Technicatome), Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)). (to CEA, 75 - Paris (France)). French Patent 2,368,122/A/. 15 Oct 1976. 13p. (In French).

The invention concerns a system for charging and discharging the fuel assemblies of a nuclear reactor, particularly although not exclusively, a liquid sodium cooled fast reactor. It lessens the drawbacks of the previous known solutions, particularly by enabling a fast fuel assembly handling rate to be sustained, whilst reducing the congestion in the transfer lock and making it possible for the latter to be adapted should the assembly guide ramps not have the same slope. To this end, the system considered includes a sealed lock capping, on the upper shield slab of the reactor, the ends of the two ramps closed by valves and respectively communicating with the reactor vessel and an external storage vessel. Inside the lock, two rail lengths carried by a vertically pivoted wheel extend the two ramps intended for receiving a mobile facility with an assembly to be charged or discharged. They slide on each ramp controlled by a winch located at the top of the lock. Each straight rail length is articulated on a horizontal shaft carried by a pivoting frame fitted in the lock and includes a lever extending the rail length beyond its shaft. This lever works in conjunction with a tilting system that moves the rail length from its inclined position in the extension of one of the ramps to the vertical position or vice versa. The invention thus combines the tilting movements of the rail lengths with a rotating movement of the wheel to redress these rail lengths and reverse their correspondence with the two ramps, in order to introduce into the vessel a new assembly simultaneously with the withdrawal from it of an irradiated assembly which, at the end of the operation, is received in a storage vessel.

- 5159 DEVICE FOR CONTROLLING FLOW RATE OF COOLANT FOR AN LMFBF TYPE REACTOR PLANT. Nakano, H.; Sato, M.; Sakuragi, M.; Iwashita, T. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-48,193/A/. 13 Oct 1976. 7p. (In Japanese).

The purpose of the invention is to minimize

the change rate and change width of outlet temperature to inlet temperature of an intermediate heat exchanger or a steam generator to decrease a thermal shock. In a plant wherein an intermediate heat exchanger is connected to a reactor through a primary piping, a steam generator is connected to the heat exchanger through a secondary piping, water is fed to the steam generator through a third piping, and the thus obtained steam is fed to a turbine, a master unit including a primary to third coolant flow setter for controlling the primary to third coolant flow is provided to minimize the change rate and change width of the outlet temperature to the inlet temperature of the intermediate heat exchanger or the steam generator, to thereby control various control means.

- 5160 SODIUM LEAK DETECTOR. Aoki, K.; Iwasaki, T. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1978-43,194/A/. 1 Oct 1976. 3p. (In Japanese).

A method to permit accurate detection of sodium leakage by distinguishing between variations in the overflow tank level due to changes in sodium temperature accompanying changes in the power level or operating state of a liquid metal sodium cooled fast breeder, and a reduction in power level due to duct leakage is described. The volume of sodium in the primary cooling system is roughly estimated from the temperatures of hot and cold legs, and the duct leak preset level with respect to the sodium liquid level within an overflow tank when the plant is normal is varied according to the state of the plant. More particularly, the volume of sodium within the overflow tank is calculated on the basis of a signal representing the liquid level detected by a liquid level gauge in the overflow tank and a signal representing the temperature detected by hot leg and cold leg thermometers, thereby obtaining the preset value of the liquid level within the tank. This liquid level preset value and the afore-mentioned liquid level signal are compared, whereby sodium leakage due to breakage of the duct or other causes can be detected independently of the power state of the plant.

- 5161 INCIPIENT BOILING OF SODIUM IN AN ANNULAR CHANNEL HEATED BY A SINGLE-PIN WITH BLOCKAGE. Fujie, Y.; Okada, T.; Miyazaki, K.; Inoue, S.; Suite, T. (Osaka Univ., Suita (Japan); Faculty of Engineering). J. Nucl. Sci. Technol. (Tokyo); 13: No. 10, 596-608 (Oct 1976).

In simulation of partial failure affecting a fast breeder core, experiments on sodium boiling were performed in a single vertical channel of annular cross section (15mm I.D.), indirectly heated by high flux heater pin (6.5mm O.D., 60 cm heating length) and provided with a blockage disk (11 mm O.D., 1 mm thick, obstructing 42% of the channel area). The experimental conditions were: Pressure of cover gas: 1.0 kg/cm² abs., Heat flux: 0 -- 80 W/cm², Flow rate of sodium: 1 -- 4 l/min. The results revealed that the degree of incipient boiling superheat is reduced to a fraction of the value obtained previously for direct joule heating under similar conditions, and that, consequently, the intensity of pressure pulses is correspondingly reduced. On the other hand, the incipient boiling superheat increases, together with the extent of its scattering, with rising intensity of the heat flux. Local boiling was observed to precede the onset of bulk boiling, causing small pressure spikes to be detected by all three pressure sensors installed along the test channel. The pressures

evaluated from the measured velocities of the liquid sodium column based on single-bubble model were compared with the measured pressure signals, and a fairly good agreement was obtained.

- 5162 DEVELOPMENT OF SODIUM PENETRATION EQUIPMENT. Nihei, T.; Miyazawa, T. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Doryokuro Giho; No. 20, 37-49 (Oct 1976). (In Japanese).

In the development of sodium-cooled fast breeder reactors, various measuring and monitoring methods are required for securing the safety and improving the operational rate, which have never been seen in conventional reactors. One of them is the requirement to monitor optically the state of the structures within a reactor as opaque liquid sodium is filled in it. The sodium penetration equipment, which scans with ultrasonic beam, monitors its reflecting or penetrating waves, and makes images, is promising. The development is carried out mainly by the Power Reactor and Nuclear Fuel Development Corporation in Japan. The main functions are the observation and identification of fuel assemblies, the observation of other structural changes, and the confirmation of the state of control rods. FFTF system in the U.S., Phenix system in France, the system for CFR in U.K., A-scope system in FRG, and PNC system in Japan are briefly explained. The sodium-penetrating equipment manufactured for trial comprises a supersonic transducer, transducer-driving mechanism, and display portion. The horizontally sodium-penetrating equipment is intended for confirming the gap between the core and the core upper mechanism, and it comprises also a supersonic transducer, transducer-driving mechanism, a reflecting plate and display portion. The basic test was carried out in water with reduced scale model, and the S/N ratio, the accuracy of fixing the reflecting plate, the limit of detecting the gap, the horizontal resolution, and mutual interference were examined.

- 5163 REPORT OF THE EXPERTS' COMMITTEE ON ADVANCED POWER REACTOR DEVELOPMENT ON MEASURES OF THE DEVELOPMENT. Genshiryoku Iinkai Geppo; 21: No. 8, 2-8 (Oct 1976). (In Japanese).

A report of the Committee on the policy of advanced reactors development in Japan, submitted to the AEC of Japan, is presented: utilization of light water reactors, establishment of Japan's own fuel cycle, development of a fast breeder reactor, development of an advanced thermal reactor, development of a multi-purpose high-temperature gas cooled reactor, introduction of overseas advanced reactors, the set up of utilizing developed reactors, and general research and development. In usage of nuclear power in Japan which has little energy resources, its development is necessary both in electric power generation and in industrial processes.

- 5164 PROVIDING RADIATION SAFETY AT NUCLEAR POWER PLANTS. Gusev, N.G. At. Energ. (USSR); 41: No. 4, 254-260 (Oct 1976). (In Russian).

A review of papers on radiation safety at atomic power stations is made. The following aspects are considered: radiation characteristics of the water coolant and their conformity with design data, personnel irradiation doses, blow-out of radioactive gases and aerosols, special gas decontamination systems, tritium at atomic power stations, radiation situation in the environment, requirements of tower-cooler layout, volume of

external radiation monitoring.

- 5165 REACTIVITY CHANGE RATE DURING THE BOR-60 REACTOR CORE LIFE. Afanas'ev, V.A.; Efimov, V.N.; Krasnoyarsov, N.V.; Skulkin, N.N.; Fedyakin, R.E. At. Energ. (USSR); 41: No. 4, 270-271 (Oct 1976). (In Russian).

ANALYTICAL SOLUTION; BOR-60 REACTOR; BURNUP; REACTIVITY; REACTOR CORES; SWELLING; REACTIVITY WORTHS

- 5166 DEVICE FOR DETECTING GASES, IN PARTICULAR FISSION GASES, IN THE COOLANT CHANNEL OF A LIQUID-COOLED NUCLEAR REACTOR FUEL ELEMENT. Gast, K. (to Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). German (FRG) Patent 1,807,801/B/. 23 Sep 1976. 3p. (In German).

In controlling sodium cooled fast nuclear reactors it is important to provide for the early detection and localization of cladding tube defects in individual fuel elements. For this reason it is proposed to install in the coolant duct of a fuel element a cyclone assigning a rotary movement to the coolant in the duct, and in the center of the cyclone a heated thermocouple as a detector. For heating purposes the thermocouple may be embedded in a body containing fissile material, or it may be heated electrically. The device according to the invention for the detection of gases is very sensitive and has a short response time.

- 5167 NEUTRON FLUX MEASURING DEVICE. Marmonier, P.; Moreau, J.; Nervi, J.C. (to CEA, 75 - Paris (France)). French Patent 2,365,186/A/. 21 Sep 1976. vp. (In French).

This invention concerns a neutron flux measuring device for liquid coolant reactors and particularly applies to fast power reactors, cooled by liquid sodium. It is essential to have facilities for permanently measuring the flux of neutrons emanating from the core, not only when the reactor starts up but also when working at full power. An improvement to the solutions already considered allows for a more accurate and more sensitive measurement of the neutron flux, even when the reactor is working at very low rates. To this effect, the device considered, comprising at least one neutron detection chamber placed under and outside the vessel and as close as possible to it, is characterized in that it includes at least a neutron guide, composed of two separate superposed components. The first, top component includes a case open at the top, having an external profile identical to that of all the core assemblies and resting through its lower part on a bearing surface of a supporting prop on the diaphragm in which penetrates a foot centrally extending the case. The second, bottom component, in which the foot of the first engages coaxially, is fixed inside the supporting prop and includes a closed tube filled with an inert gas, extending past the diaphragm to near the internal surface of the bottom of the vessel. The first component contains a neutron emitting medium extending at least over its major part above the mid plane of the core.

- 5168 METHOD FOR LEAK-TESTING OF LIQUID-METAL WATER/STEAM HEAT EXCHANGERS. Dumm, K. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). German (FRG) Patent 2,329,549/C/. 16 Sep 1976. 5p. (In German).

A method is proposed for leak-testing of liquid-metal water/steam heat exchangers before

their initial start-up being applied at pressures and temperatures corresponding to those of later working conditions. The invention provides for the heat exchangers to be acted upon by a testing gas at testing pressure on the water/steam side and by hot liquid-metal on the liquid metal side. The cover gas atmosphere (argon) on the liquid metal side is monitored for the appearance of the testing gas which is present on the other side. This way, corrosion and the resulting repairs are avoided in the operation of nuclear power plants with sodium cooling.

- 5169 NUCLEAR REACTOR WITH ROTATING CLOSURE. Mohm, F. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). German (FRG) Patent 1,564,114/C/. 16 Sep 1976. 7p. (In German).

The invention concerns the arrangement of the decay channels of a nuclear reactor which has a closure which is eccentrically rotatable around its middle axis. According to the invention, the decay channels lie on the circle produced by the refuelling channel on rotation of the closure, but at another position than the charging channels, within the shielding. Therefore, the shielding may be used as decay store for the spent fuel elements and the time necessary for refuelling may be shortened. A rotating auxiliary cover facilitates the discharging procedure.

- 5170 ELECTROCHEMICAL SYSTEM FOR DETERMINING THE OXYGEN CONCENTRATION IN A LIQUID ALKALINE METAL. Roy, P.; Licina, G.J. (to General Electric Co., Schenectady, NY (USA)). French Patent 2,325,928/A/. 16 Sep 1976. 10p. (In French).

A description is given of an electrochemical system for determining the oxygen concentration in a liquid alkaline metal. Its features include: (a) a solid electrolytic body conducting oxygen-ions having a pair of opposed lateral surfaces, one being adapted so as to be in close contact with the liquid metal whose concentration in oxygen it is desired to determine; (b) a mixture containing a known concentration of one of the metals selected from among gallium, indium or tin and an oxide of this metal. This mixture is liquid at the operating temperature of the system and is in close contact with the second lateral surface of the electrolytic body so as to provide a reference electrode, and (c) an instrument for measuring any electromotive force generated between the reference electrode and the liquid metal by conduction of the oxygen-ions through the solid electrolyte in order to provide a reading indicating the oxygen concentration.

- 5171 CONTROL DEVICE IN A REACTOR. Onuki, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-34,094/A/. 9 Sep 1976. 2p. (In Japanese).

The purpose of the invention is to flatten temperature distribution of coolant within a core. The control device of the present invention is to vary reactivity of a fast breeder to control a reactor power. In general, the control device of this kind comprises a guide pipe arranged within the core and a control rod movable up and down within the guide pipe, and a coolant flows from bottom toward top within the guide pipe. Since a cooling flow rate has a margin, temperature of coolant outlet is extremely low as compared to a fuel assembly, and therefore temperature gradient in the vicinity of the top of the control rod becomes sharp to possibly impart

thermal shock to the structural material. In the present invention, the flow passage of coolant is varied to thereby avoid outflow thereof into the core, thus flattening the temperature distribution of the coolant within the core.

- 5172 TESTING N 205 TYPE ANTIFRICTION BEARINGS IN LIQUID SODIUM. Metal, D. (Vyzkumny Ustav Zavodu Energetického Strojirenstvi, Brno (Czechoslovakia)); Handl, Z. (Vyzkumny Ustav pro Valiva Loziska, Brno (Czechoslovakia)). Strojirenstvi; 26: No. 9, 554-558 (Sep 1976). (In Czech).

Comprehensive tests were carried out of special N 205 antifriction bearing in liquid, hot sodium at temperatures of 370 degC and 420 degC. Sodium used for tests contained various amounts of oxygen. Visual inspection, X-ray spectral microanalysis, X-ray phase analysis and accurate measurements before and after tests were used to assess the influence of hot aggressive metal. It was found that under the conditions simulated by the tests the ball bearings can be expected to have a short service life.

- 5173 FUEL ASSEMBLY. Ito, A. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-29,495/A/. 31 Aug 1976. 5p. (In Japanese).

The purpose of the invention is to improve thermal efficiency and safety of a FBR type reactor and prevent the generation of cavitation by permitting a desired quantity of coolant to flow into a fuel assembly used in the reactor. A fuel assembly accommodating a bundle of fuel rods is provided at its lower end with cylinders coaxial with respective fuel rods by respective connecting tubes, and a flow rate control mechanism is provided to cause the coolant to flow toward the inner wall of each of the cylinders. The flow rate control mechanism consists of, for instance, a guide hole formed in each cylinder and a round rod fitted in the associated connecting tube.

- 5174 POSITION CONTROL DEVICE FOR A CONTROL ROD. Ono, T.; Kusaka, S. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-29,498/A/. 31 Aug 1976. 6p. (In Japanese).

The purpose of the invention is to reliably prevent dangerous operation in the control of the position of the control rod by checking for abnormal pulse motor coil excitation voltage and, at the time of occurrence of abnormality, immediately holding the control rod stationary lest it should be moved to an unsafe position, this being accomplished excitation from a compensating excitation system. In an LMFBR reactor, a circuit for memorizing the correct output states of individual drive signals at arbitrary instants and consequently producing the memorized results is provided, and the output of the circuit and the actual drive signal are compared at all times to discriminate whether the drive signal being compared is normal or not. When the actual drive signal is abnormal, a series signal varying after a predetermined pattern is shifted to enable replacement of the actual drive signal, so that irrespective of whether the problem drive signal is "on" or "off", a drive signal of the correct pattern may be supplied to the pulse motor to hold the control rod and prevent it from being moved toward the dangerous side due to its own weight or other causes.

- 5175 SODIUM LEAK DETECTOR. Tsukamoto, M.; Ibe, H. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1978-22,994/A/. 13 Aug 1976. 4p. (In Japanese).

The object of the invention is to detect change inflow due to clogging of a filter while maintaining a pressure differential, between upstream and downstream of the filter, constant, to enable quick responsive leak detection in a simple device. The atmosphere gas at a sampling point is subject to sampling by a blower via a filter and a flow meter. The pressure gage measures gas pressure in a sampling tube and controls the number of revolutions of the blower so that the gas pressure is constant. This gas pressure may be maintained constant to thereby maintain a pressure difference between the upstream and the downstream of the filter for each filter. A flow signal of the flow meter is transmitted to a signal processor via a multiplexer, and when the change rate of flow with respect to an initial flow reaches a threshold value, and alarm is given to indicate that sodium leak occurs.

- 5176 HEAT TRANSFER SYSTEM IN A FAST BREEDER. Suzuoki, A.; Sagawa, N. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1978-21,393/A/. 11 Aug 1976. 4p. (In Japanese).

The object of the invention is to remove a radioactive corroded product contained in liquid metal in a primary cooling system to decrease a dose rate of surface radiation of the machineries and pipes in the primary cooling system. There is provided a pressure-reduced distilling device having liquid metal on the primary within a flow passage of a primary cooling system used as a heating source and having fluid on the secondary within a flow passage of a secondary cooling system used as a cooling source, and the liquid metal on the primary containing impurities is partly introduced into the pressure-reduced distilling device, within which the impurities are removed, and the liquid metal on the primary with the impurities removed is returned into the flow passage of the primary cooling system.

- 5177 SAFETY EQUIPMENT IN A REACTOR. Shiratori, H.; Ishiyama, S.; Ugawa, Y. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-21,391/A/. 10 Aug 1976. 3p. (In Japanese).

The object of the invention is to safely retain, even if fuel should be molten and flown through the bottom of a container in a reactor, the molten fuel to remove heat generation of the fuel to prevent occurrence of a critical trouble. A reactor container housing a core and coolant has thereunder a separation dome in a central portion thereof and a partitioning plate coaxially and circularly disposed in the periphery of the separation dome, with a tray formed of magnesium oxide being disposed. Further, a cooling path system is provided so as to surround the tray. The cooling path system and the reactor container are surrounded and protected by a reactor wall provided with heat insulating refractory bricks, a coolant pouring system extends through the reactor wall, and the coolant is supplied to the tray.

- 5178 RADIATION SHIELDING DEVICE. Matsumoto, F.; Yamazaki, H. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-20,099/A/. 9 Aug 1976. 2p. (In Japanese).

The object of the invention, particularly in a fast breeder, is to positively shield

radiation leaking from a portion through which a measuring cable extends, the cable extending from the interior to the exterior of the reactor. An inner extension pipe connected to a drive portion of a fuel assembly is formed with a through hole, through which a measuring cable from a detector guides externally of the reactor. The support plates are provided on inlet and outlet of the through hole, interiorly of which is filled with boron carbide (B₄C) powder to absorb neutron to prevent leakage of radiation.

5179 PROPOSED STRATEGY FOR POWER REACTOR DEVELOPMENT LETS P.N.C. DOWN. At. Jpn.; 20: No. 8, 6-11(Aug 1976).

The strategy for power reactor development in Japan is discussed. The development of nuclear power is at the stage of requiring larger funds than ever. The developments of the prototype heavy water reactor Fugen and the experimental fast reactor Joyo are at the point of planning the next stages. The high temperature gas reactor project must be forwarded. The fuel cycle picture is a serious problem in Japan, and the national project on the development of heavy water reactors was started, and has been developed by the Power Reactor and Nuclear Fuel Development Corporation (PNFC). However, the users criticized this project. The prototype heavy water reactor will be in operation next year, and it will be after the prototype construction, switching should be made to a thorium-fueled reactor. Discord in the basic strategy for power reactor development arises from a plan to introduce CANDU reactors.

5180 SOWACS-3: A CODE FOR ANALYZING THE SYSTEM EFFECTS OF LARGE-LEAK SODIUM-WATER REACTIONS IN STEAM GENERATORS FOR LMFBR'S. Maekawa, I.; Shindo, Y. (Kawasaki Heavy Industries Ltd., Kobe (Japan)). FAPIG (Tokyo); No. 82, 31-42(Aug 1976). (In Japanese).

In liquid metal-cooled fast breeder reactors, since heat exchange is performed between sodium and water or steam, sodium-water reaction occurs if sodium leak arises, and it may give great influence on the secondary cooling system. Large leak sodium-water reaction shows the complex phenomena related to hydrodynamics, thermodynamics, and structural dynamics, and more complicate behavior is shown because the transient phenomena of the release system are superposed. SOWACS-3 (Sodium Water reaction in Compressible Sodium-3) is a code to analyze the instant complex phenomena in the large leak of steam generators, and the effect of sodium-water reaction over secondary coolant pipes, intermediate heat exchanger, steam generator and release system is analyzed. SOWACS-3 analyzes the one dimensional model of unsteady compressible fluid in accordance with flow equations and by the characteristics curve method. The report describes the fundamental formulae, numerical calculation procedures and the construction of computing codes.

5181 METHOD OF PREHEATING A NUCLEAR REACTOR COOLING SYSTEM. Kurokawa, M. (to Mitsubishi Heavy Industries Ltd., Tokyo (Japan)). Japanese Patent 1978-16,196/A/. 29 Jul 1976. 4p. (In Japanese).

The object of the invention is to permit efficient and uniform preheating of the cooling system of a nuclear reactor by utilizing apparatus constituting the cooling system, particularly utilizing an overflow column, through which preheating gas is supplied and discharged. When preheating a cooling system before the pouring of liquid sodium, a

cylindrical partition member is temporarily inserted into an overflow column with its lower end aligned with the outlet end of the overflow column, thus dividing the inner space in the column into a supply passage for supplying preheating gas and an exhaust passage for exhausting the preheating gas. A preheating gas supply source is connected to the upper end of the supply passage. Circulation of the preheating gas proceeds as follows. The preheating gas having returned after excursion through the cooling system in the reactor is exhausted from the exhaust passage and returned to the preheating gas supply source, and after re-heating it is supplied again to the cooling system.

5182 ABNORMAL CONDITION DETECTOR OF A REACTOR CONTAINER. Kanekubo, H. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-13,097/A/. 22 Jul 1976. 4p. (In Japanese).

The object of the invention is to readily and accurately detect abnormality and abnormal point of a lining applied on the inside of a concrete wall within a container in a sodium cooled fast breeder. A dual lining made of steel is disposed internally of a room surrounded by concrete to house cooling equipment and piping therein. A space formed between the dual lining is divided into a plurality of spaces by partitioning plates. Each space is formed with an opening, which is connected to a pipe, and connected to a blower in an annulus exhaust system through a respective opening and closing valve in the midst thereof. A leak detector is provided in the midst thereof. The space in the lining is always maintained at negative pressure. When the abnormality is detected, the aforesaid valves are all closed and subsequently these valves are opened one by one. This operation can tell which divided space has abnormality.

5183 REFUELING MACHINE. Ito, M.; Ugawa, S.; Ito, T. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-11,296/A/. 20 Jul 1976. 4p. (In Japanese).

The object of the invention is to prevent occurrence of such a trouble, when fuel assembly is disposed, that the fuel assembly to be disposed tends to interfere with another fuel assembly adjacent thereto resulting in failure of disposition. Signals of an angle sensor for sensing a relative angle between a rotary plug and a rotary housing and of an angle sensor for sensing a relative angle between the rotary plug and a fixed plug may be processed by an adder to sense a relative angle between a gripper arm and a core. When the fuel assembly is removed, a differential angle α relative to a reference direction may be sensed by the adder to memorize it by means of an angle displacement detector. On the other hand, a differential angle β relative to a reference direction, when the fuel assembly is disposed, may also be sensed by the adder similarly to the former to thereby sense a differential theta with respect to the α previously memorized in the angle displacement sensor by an operation. Where the theta is close to 30°, it means that performance of fuel assembly disposition is dangerous.

5184 INVERSE STEAM GENERATOR ESPECIALLY FOR LIQUID METALS. Krizek, V.; Dubsek, F. Czech Patent 163,989/B/. 15 Jul 1976. 3p. (In Czech).

The invention claims an inverse steam generator for liquid metals whose heat exchange

tubes have a non-circular (hexagonal, elliptic, cruciform, lemniscate) cross-section and are arranged in a parallel bundle.

- 5185 RAPID-ACTION VALVE ESPECIALLY FOR LIQUID METAL. Velek, V.; Cejka, J.; Jakl, J. Czech Patent 163,608/B/. 15 Jul 1976. 5p. (In Czech).

A rapid-action large-diameter valve and control fitting was designed for cooling circuits of nuclear power plants with fast liquid metal cooled reactors, namely sodium cooled. The design meets the requirements for axial symmetry about the flow axis and secures the equilibrium of hydraulic and hydrodynamic forces observing the uniformity of the thickness of the fitting walls which is a prerequisite for good resistance to thermal shock.

- 5186 IMPROVEMENTS TO LEVEL INDICATORS FOR LIQUID METALS. Playfoot, K.C.; Todd, W.H. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). French Patent 2,318,409/A/. 12 Jul 1976. Priority date 15 Jul 1975, United States of America (USA), 9p. (In French).

An inductive apparatus is described for detecting the level of a liquid metal, sodium for instance. It includes an inductive probe that may be introduced into a sealed tube directed downwards in the liquid metal chamber, comprising a twin-wire transformer; a constant amperage electricity supply electrically connected to the primary winding of the twin-wire transformer; an amplifier electrically connected to the secondary winding and to the compensating coil, the output signals of these windings being applied to the amplifier in order to produce an adjusted temperature output signal indicating the depth of liquid metal around the probe.

- 5187 FUEL ASSEMBLY IN FBR TYPE REACTOR. Miki, K. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1978-6,795/A/. 9 Jul 1976. 4p. (In Japanese).

The object of the invention is to make a wall thickness of a cladding tube at the top of fuel pins in the outermost periphery thicker than that of other portions so as to eliminate fretting corrosion of the cladding tube at the top of fuel pins when the latter are curved and deformed. A ring spacer is attached to the top of a fuel pin so as to increase the wall thickness of a cladding tube at the top portion. This prevents section deformation of the fuel pin due to outside load and avoid direct contact between the cladding tube and a wrapper tube wall at the top of fuel pin in the outermost periphery to prevent fretting corrosion of the cladding tube.

- 5188 REACTOR VESSEL. Wade, E.E. (to Westinghouse Electric Corp., Pittsburgh, Pa. (USA)). Swedish Patent 2,555,270/A/. 8 Jul 1976. 17p. (In German).

In the case of reactor vessels, especially in that of LMFBR type reactors, difficulties arise in lubricating the bearings and seals for the ducts in the top of the vessel. According to the invention, this is to be avoided by special supporting members which receive the rotatable parts of the cover and the design of which prevents the lubrication from entering into the inner part of the reactor.

- 5189 NUCLEAR FUEL ASSEMBLY. Ito, A.; Wakamatsu, M. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-5,389/A/. 6 Jul 1976. 4p. (In

Japanese).

The object of the invention is to permit the coolant in an LMFBR type reactor to enter from the entrance nozzle into a nuclear fuel assembly without causing cavitation. In a nuclear fuel assembly, which comprises a number of thin fuel pins bundled together at a uniform spacing and enclosed within an outer cylinder, with a handling head connected to an upper portion of the outer cylinder and an entrance nozzle connected to a lower portion of the cylinder, the inner surface of the entrance nozzle is provided with a buffer member and an orifice successively in the direction of flow of the coolant. The coolant entering from a low pressure coolant chamber into the entrance nozzle strikes the buffer member and is attenuated, and thereafter flows through an orifice into the outer cylinder.

- 5190 LIQUID METAL COOLED NUCLEAR REACTOR. Durston, J.G.; Hind, J.R. (to UKAEA London Office). French Patent 2,317,734/A/. 6 Jul 1976. Priority date 7 Jul 1975, United Kingdom of Great Britain and Northern Ireland (UK), 11p. (In French).

The nuclear reactor described is cooled by a liquid metal. It includes a reactor containment filled with a mass of liquid metal in which is immersed a core enclosed in a vessel. It comprises a system to cover the internal surface of the vessel wall with liquid metal taken from the area of the mass outside the vessel and to circulate the liquid metal from there radially internally towards the core.

- 5191 FAST NEUTRON REACTOR. Jogand, P. (to CEA, 75 - Paris (France)). French Patent 2,357,987/A/. 6 Jul 1976. 9p. (In French).

The invention relates to a fast nuclear reactor of the integrated type in which the hot sodium on leaving the core, contained in the internal vessel, is made to flow between the core and the side wall of this vessel to be removed from it by manifolds connected together at the bottom of each exchanger with a central channel provided in it. Under this invention, the residual power can be evacuated, in the event of reactor trouble, by causing the liquid metal to flow by natural convection. To this effect, the upper end of the internal vessel is capped with an annular bell inverted and placed under the internal pressure of a neutral gas to prevent, in normal operation, direct communication between the hot liquid metal inside the internal vessel and the cold liquid metal in the annular space between the main vessel and the internal vessel. This bell caps an auxiliary exchanger outwardly circling the internal vessel in such a way that, in the event of reactor trouble and after the pressure of the neutral gas under the bell has been reduced, a flow of the liquid metal under the bell of the internal vessel takes place by natural convection towards the annular space by passing through the auxiliary exchanger.

- 5192 DEVICE OF EXTINGUISHING A LIQUID FUEL BY REMOTENESS. Colome, J.; Duchene, A.; Regnier, J. (to CEA, 75 - Paris (France)). French Patent 2,357,267/E/. 5 Jul 1976. 8p. (In French).

In the main patent specification a description was given and a claim made with respect to a system for the rapide extinction of an inflamed liquid combustible substance by isolating it rapidly from the ambient air. This system applies; inter alia, to a sheet of sodium flowing at high temperature accidentally coming from a nuclear installation and

particularly a fast nuclear reactor. This addendum concerns an improvement to the arrangements explained in the main specification, making it particularly possible to increase by any proportion the total capacity available for collecting the inflamed liquid by means of a receptacle formed of several separate tanks, fitted with appropriate connections permitting communication between these tanks to equalise the level of the liquid fuel collected in them, on the communicating vessels principle.

- 5193 PIPE SUPPORT FOR USE IN A NUCLEAR SYSTEM. Pollono, L.P.; Mello, R.M. (to Westinghouse Electric Corp., Pittsburgh, Pa. (USA)). French Patent 2,317,583/A/. 2 Jul 1976. Priority date 9 Jul 1975, United States of America (USA), 17p. (In French).

Priority claim: 9 Jul 1975, US.

Description is given of a vertical pipe support system. It comprises a tubular pipe support structure having the same inside diameter and the same wall thickness as the pipe, the pipe support structure having a generally triangularly shaped extension formed integral with and extending circumferentially around its outward side, the bottom side of this extension generally forming a ledge; an annular load-bearing insulation formed adjacent to the extension; means for clamping the load-bearing insulation to extension; and means for providing constant vertical support to means for clamping.

- 5194 DEVELOPMENT OF COLD TRAPS. Mochizuki, K.; Nagase, N. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Doryokuro Gihō; No. 19, '37-45 (Jul 1976). (In Japanese).

Cold traps for liquid sodium coolant systems are described. Three basic designs of cold traps used at Oarai Engineering of Power Reactor and Nuclear Fuel Development Corp. are shown. Impurities are trapped by the cold traps at the entrance and exit of the mesh portion. The concentration of oxygen in liquid sodium depends on temperature. The operation temperature of cold traps for the experimental reactor Joyo is set at 150 degree C. In the secondary cooling system, hydrogen concentration is more important, and the temperature is decided by this concentration. It is necessary for designing a fast reactor to estimate the amount of impurities accumulating during its life time. In a primary cooling system, oxygen in purchased sodium, initially residual oxygen in the system, oxygen adsorbed on the surfaces of structural materials, oxygen brought in with fuel, oxygen transferred from cover gas and seal gas, and oxygen entering at the time of maintenance are considered. In a secondary system, Oxygen and hydrogen must be taken into account. Efficiency of trapping and the amount of trapped oxygen are described. Further improvement of cold traps is continued.

- 5195 SODIUM-WATER REACTION TEST FACILITY (SWAT-3). Shimazu, H.; Ukechi, K.; Sasakura, K.; Kusunoki, J. (Ishikawajima-Harima Heavy Industries Co. Ltd., Tokyo (Japan)). IHI Eng. Rev.; 9: No. 3, 81-92 (Jul 1976).

In the development of the liquid metal cooled fast breeder reactor (LMFBR), the steam generator (SG) is considered one of the most important components. The Power Reactor and Nuclear Fuel Development Corporation (PNC) is now promoting the research and development of the SG system used with the prototype fast breeder reactor 'Monju'. In this research, the phenomena of the sodium-water reaction in the SG are the key which must be investigated

for the solution of problems. The test facility (SWAT-3) simulating Monju's SG on the scale of 1/2.5 was designed, fabricated and installed by IHI at Oarai Engineering Center of PNC, its pre-operation being accomplished in February 1975. The purpose of SWAT-3 is summarized as follows: (1) To perform an overall test on the safety of Monju's SG and intermediate heat transport system under the design condition against sodium-water reaction accidents. (2) To investigate the damage of the SG structure caused by the sodium-water reaction, and the possibility of repair and recovery operations. The first test was accomplished successfully on June 9, 1975. As a result of the test, the fundamental function of this test facility was proven to be satisfactory as expected.

- 5196 APPARATUS FOR PURIFYING A PRIMARY COVER GAS. Arai, S.; Yoshizawa, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-4,199/A/. 30 Jun 1976. 4p. (In Japanese).

Object: To prevent fine liquid metal particles and fission products floating in a primary cover gas from depositing upon a shield plug and in the surrounding space in an LMFBR type reactor. Structure: In a closed loop constituted by a reactor container provided with a shield plug and supporting the core top mechanism in a fixed position, a means for eliminating or reducing impurities and radioactivity in the gas and gas circulating means, these means being provided outside the container, and duct lines valves connecting the container and these means, a nozzle header having a plurality of nozzles is provided at underside of the shield plug and connected thereto by a duct line. The primary cover gas which is purified and fed out of the container is supplied to the nozzle header and sprayed therefrom to form an air curtain. This air curtain can prevent intrusion of the cover gas into the afore-mentioned space.

- 5197 MAINTENANCE EQUIPMENT FOR A FAST REACTOR. Yoshizawa, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-4,189/A/. 30 Jun 1976. 3p. (In Japanese).

The object of the invention is to permit cleaning of through-holes of various diameters by mounting a spray nozzle on the lower end of a vertically movable extension tube connected to an Ar gas heater. After removing control rod drive mechanism and other apparatus mounted on the top of the reactor core for maintenance, a spray nozzle of a maintenance apparatus on a reactor top pit lid is inserted into a through-hole in a shield plug. Then, Ar gas heated by the heater is supplied through the extension tube and sprayed, thereby removing Na slug attached to and solidified on the inner surface of the hole by fusion.

- 5198 FUEL ASSEMBLY. Wakamatsu, M.; Ikeda, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-1,787/A/. 29 Jun 1976. 5p. (In Japanese).

Object: To provide a fuel assembly having a handling head connected at the upper part thereof and an entrance nozzle connected at the lower part thereof, said assembly comprising spiral protrusions provided in the outer peripheral surface in the vicinity of a coolant inflow end of the entrance nozzle to restrain an inflow quantity of coolant into a low pressure cooling chamber. Structure: Spiral protrusions are provided in the outer peripheral surface adjacent the end of the

entrance nozzle in which coolant flows in to limit gaps between the fuel assembly and the partitioning wall and connection tube thereby controlling the flow of coolant into the low pressure coolant chamber so that even when the fuel assembly is eccentrically deformed, the flow distribution of the coolant in the blanket region can be maintained properly.

- 5199 NUCLEAR REACTOR. Ito, A.; Kumaoka, Y. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-1,793/A/. 28 Jun 1976. 3p. (In Japanese).

The object of the invention is to improve the safety of an LMFB type reactor and also increase the thermal efficiency thereof by reducing the flow of the coolant from high pressure side to low pressure side even when floating of the fuel assembly results. A labyrinth is formed on the inner surface of a duct connecting a lower low pressure coolant chamber and an upper low pressure coolant chamber or on the surface of a lower portion of the fuel assembly inserted in the duct. With this labyrinth the rate of the coolant passing through a contact section between the fuel assembly and an upper partition member within the reactor is controlled.

- 5200 CONTROL ROD DRIVE MECHANISM. Yamada, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1978-1,792/A/. 28 Jun 1976. 3p. (In Japanese).

The object of the invention is to reduce dislocation of upper and lower guide tubes for a control rod from their common axis in an LMFB type reactor using a rotary plug for fuel exchange and also prevent adverse effects upon the operation of the rotary plug. The upper guide tube has a lower tapered open portion, which is adapted to be inserted in the lower guide tube during the normal operation of the reactor. It is adapted to be released from the lower guide tube by raising the entire control rod drive mechanism. Since the upper and lower guide tubes are coupled to each other by insertion, as mentioned above, even slight dislocation is caused by an earthquake or the like, they will remain coupled to ensure stable scram operation.

- 5201 SYSTEM OF PENETRATION OF A NUCLEAR REACTOR SLAB FOR MEASUREMENT CABLES. Eyral, M.; Mahe, A. (to CEA, 75 - Paris (France)). French Patent 2,357,037/A/. 28 Jun 1976. 13p. (In French).

This invention concerns a device that enables the cables, connecting the sensors located inside a nuclear reactor vessel to the machines outside the vessel processing the information picked up by these sensors, to go through the vessel in a leak-tight manner. The aim of this device is to facilitate, by special means, the fitting of the sealing device through the upper shield and the installation of the cables when the nuclear reactor, of the fast liquid sodium cooled type, is being assembled. It includes a vertical tube closed at its lower end by a removable cover and at its upper end by a flange connecting it to the upper shield, bored to ensure a tight fit with respect to the upper shield, and a removable closure capping located above the flange. The tube is internally provided with at least one plate perpendicular to the axis of the tube, connected in a leak-tight manner to the tube by its periphery, the plate being fitted with holes for the leak-tight passage of a cable with a special insulating sheathing.

- 5202 EXTREMELY SENSIBLE METHOD AND DEVICE FOR DETECTING METALLIC VAPORS. Sun, K.H. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). French Patent 2,316,597/A/. 25 Jun 1976. Priority date 2 Jul 1975, United States of America (USA), 12p. (In French).

A method is described for detecting the presence of a metal gas in an enclosed atmosphere by means of a radioactive labelling substance. The metal gas comprises a metal selected from sodium, potassium and their mixtures. A current of this atmosphere is sent into an exchange column containing this labelling substance, then through an ionisation chamber to achieve the detection of the radioactivity and hence provide an indication of the presence of a metal gas. The characteristic of this method is that this radioactive labelling substance is capable of freeing tritium when it is in contact with the metal gas and includes at least one chemical compound in the group formed by XT_n , where T is the tritium and X is -OH, -OR, RCOO-, RO- and their mixtures, with R being equal to $C_{sub}(n)H_{sub}(2n+1)$ and n included between 1 and 20 inclusive.

- 5203 PROCESS AND SYSTEM FOR DETECTING AND PRE-LOCATING A CLAD FAILURE IN A FAST NEUTRON NUCLEAR REACTOR. Paziaud, A. (to CEA, 75 - Paris (France)). French Patent 2,356,244/A/. 23 Jun 1976. 6p. (In French).

The invention relates to a process and system for detecting and pre-locating a clad failure in the fuel assembly of a fast nuclear reactor cooled by a liquid metal (generally sodium) by counting the delayed neutrons emitted by the fission products escaping from the failed assembly and taken up by the liquid metal. The invention makes it possible to increase sensitivity in the detection of the neutron coming from the fission products. To this effect the process consists in performing a direct measurement of the delayed neutrons in the reactor coolant flow, inside the vessel itself containing this coolant. The measurement device has at least one detector for the neutrons emitted by the fission products carried in the coolant flow as a result of the failed fuel element. This detector is inserted inside the vessel in the coolant flow in an area protected from the neutrons emerging from the core by the coolant volume itself and by structures immersed in the annular space between the core and vessel. Preferably, the neutron detector is formed of a U-235 fission chamber, fitted inside a thimble immersed in the coolant.

- 5204 COOLING DEVICE OF A SPENT FUEL. Fujita, M. (to Fuji Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-154,996/A/. 18 Jun 1976. 4p. (In Japanese).

The object of the invention is to provide a bypass pipe between a coolant outlet pipe and a coolant inlet pipe to prevent occurrence of siphon effect, thus providing fail-safe means. In normal operation, sodium circulates from an outlet pipe to an inlet pipe through a pump and a cooler, while a part of sodium passes through a bypass pipe to be short-circuited. While the short-circuited flow is restricted by a throttle, it may be assured by a flow meter that the bypass pipe is not blocked. On the other hand, if a rupture should occur at a lower point of the pipe, the sodium within a spent fuel storage tank would be flown out because of the siphon effect of the inlet pipe and outlet pipe with the result a liquid level lowers. However, when the liquid surface is

exposed from the outlet pipe, then cover gas is sucked therethrough and also into the inlet pipe passing through the bypass pipe, whereby the siphon effect is impaired.

- 5205 LIQUID METAL COOLED REACTOR. Winnett, G.F. (to UKAEA London Office). French Patent 2,316,704/A/. 18 Jun 1976. Priority date 20 Jun 1975, United Kingdom of Great Britain and Northern Ireland (UK), 11p. (In French).

A liquid metal cooled reactor is described. The reactor core is immersed in the liquid metal contained in a primary vessel located in a containment concrete vessel. The upper part of the concrete vessel is insulated by a series of plates.

- 5206 FUEL ASSEMBLY FOR A FBR TYPE REACTOR. Kakehi, I.; Matsuzuka, R. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-154,987/A/. 16 Jun 1976. 4p. (In Japanese).

The object of the invention is to allow a fuel assembly to be oriented by the assembly itself when the latter is charged into a core matrix, thus providing a smooth charging. A bearing is secured to a tapered surface disposed slightly lower than a level at which an outer cylindrical pipe in a tapered surface at an upper end of a handling head having a polygonal section or in an upper part of an entrance nozzle. With this structure, at the time of charging, one fuel assembly comes into contact with the other fuel assembly previously charged through a ball bearing, whereby the assembly is received in position while being rotated smoothly.

- 5207 SAFETY EQUIPMENT OF A REACTOR. Kasama, S. (Nippon Atomic Industry Group Co. Ltd., Tokyo). (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-154,988/A/. 16 Jun 1976. 2p. (In Japanese).

The object of the invention is to retain melting fuel, when a reactor vessel is failed as a result of melting a core, to remove heat generation of fuel, thus preventing recriticality accident. The safety equipment, which has a receiving vessel and a cooling device disposed below a reactor protector vessel, provides an arrangement wherein the melting fuel, which has led to failure of the reactor vessel and the protective vessel resulting from melting trouble of a core, drops onto the receiving surface of the receiving vessel. The thus dropped melting fuel is dispersed on the receiving surface by the action of protrusions formed on the receiving surface, thereby preventing the recriticality accident. The melting fuel gradually dissolves UO_2 or MgO which constitute the receiving surface. Sodium is fed through a sodium feed pipe, the sodium being heated by decay heat of the melting fuel and then boiled and vaporized. The decay heat of the melting fuel is partly removed even by the vaporized heat.

- 5208 APPARATUS FOR RETAINING MOLTEN FUEL IN A REACTOR. Sato, Y. (Nippon Atomic Industry Co. Ltd., Tokyo). (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-15,3097/A/. 16 Jun 1976. 4p. (In Japanese).

The object of the invention is to retain and cool molten fuel, when a reactor vessel is failed, to avoid assumption of a recritical state, thereby securing safety of the reactor. A molten fuel retaining apparatus formed with a

molten fuel cooling tray in an outer peripheral surface of a hollow conical body, which is internally formed with coolant passages and has a plurality of coolant passing holes, is disposed below a reactor vessel within a primary tank which surrounds the reactor vessel housing therein a primary coolant for cooling the core.

- 5209 NUCLEAR REACTOR CONSTRUCTION COOLED WITH LIQUID METAL. Barnes, S.; Fothergill, J.R. (to UKAEA London Office; Deutsches Patentamt, Muenchen (Germany, F.R.)). German (FRG) Patent 2,555,245/A/. 16 Jun 1976. 13p. (In German).

A liquid metal cooled reactor is presented possessing a device to transmit ultrasonic signals into a melt bath of the coolant and out of the latter by means of a liquid wave-guide consisting of a column of reactor coolant which is not interrupted by a membrane.

- 5210 DEVICE FOR THE TRANSFER OF COOLANT IN A NUCLEAR REACTOR. Mangus, J.D. (to Westinghouse Electric Corp., Pittsburgh, Pa. (USA)). French Patent 2,314,561/A/. 9 Jun 1976. Priority date 10 Jun 1975, United States of America (USA), 14p. (In French).

A system is described for the circulation of a nuclear reactor coolant, comprising a reactor vessel, a heat exchanger and a coolant circulation pump interconnected by pipes so that the coolant circulates in turn, by means of the pump, through the reactor vessel and the heat exchanger.

- 5211 SPENT FUEL CLEANING DEVICE. Sasaki, S. (to Tokyo Shibaura Electric Co. Ltd.). Japanese Patent 1977-148,799/A/. 4 Jun 1976. 3p. (In Japanese).

The object of the invention is to clean spent fuel of a fast breeder reactor safely and positively by a simple procedure. At the bottom of a cleaning tank is provided a carbon dioxide gas feeding device filled with carbon dioxide and a cleaning solvent tank for feeding lower pressure vapor is connected thereto. The spent fuel is put into the cleaning tank to clean it so that sodium adhered to the surface of the spent fuel may be cleaned while effecting reaction expressed by $2Na + 2CO_2 + 2H_2O \rightarrow 2NaHCO_3 + H_2$. On the other hand, hydrogen produced is discharged through a waste processing system, and concentration of hydrogen is maintained outside explosion of hydrogen concentration by means of dilution of rare gases.

- 5212 SAFETY EQUIPMENT IN A REACTOR. Nidome, J.; Kuroki, A. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-148,792/A/. 4 Jun 1976. 5p. (In Japanese).

A method to satisfactorily protect storage equipment such as a primary tank, chamber walls of a reactor and the like even in case of trouble such as fuel melt, to prevent scattering of radioactive material to enhance safety of the reactor is described. A molten fuel holding device disposed within a molten fuel holding vessel and having a molten fuel cooling tray disposed in an outer periphery of a hollow conical body formed with a plurality of cooling water passing holes has its tray to receive molten fuel extending through a reactor container in scattered relation in the entire peripheral direction, and coolant, which enters through a hollow portion of the hollow conical body and outflows through the coolant passing holes to be heat-exchanged with coolant in the

primary tank for natural circulation, is used for cooling.

5213 FY 1976 OPERATION PROGRAM OF PNC. At. Jpn.; 20: No. 6, 29-36(Jun 1976).

The operational program of Power Reactor and Nuclear Fuel Development Corporation for FY 1976 is amended as follows. The development of power reactors, amounting to 16,368 billion (government subsidy) and 38,829 billion, covers the development of the fast breeder reactors, JOYO and MONJU, the development of the advanced thermal reactor ATR (heavy water reactor), the operation and preparation of equipments and facilities for common purposes, and the planning and management of research and development. The work on the experimental reactor JOYO will be advanced with the continuation of integral performance tests aiming at reaching criticality by the end of 1976. The sector experiments of the prototype reactor core will be carried out in the fast critical assembly (FCA) of the Japan Atomic Energy Research Institute (JAERI). PNC plans to participate in the BIZET experiment program, designed jointly with West Germany and the UK. The criticality of the prototype ATR will be attained in 1977. The facilities for plutonium fuel development and fuel material inspection will be introduced. Development of nuclear fuel, amounting to 2,303 billion (government subsidy) and 11,137 billion, covers the overseas surveys, the domestic prospecting in the Tohno district, the research at Ningyo-Toge mine, the utilization of plutonium, nuclear fuel cycle, and the development of uranium enrichment by centrifuges and cascades. The reprocessing of spent fuel amounting to 2,804 billion (government subsidy) and 13,985 billion, covers the reprocessing facilities and the development of techniques.

5214 SAFETY EQUIPMENT FOR A REACTOR.

Nagashima, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-144,589/A/. 26 May 1976. 4p. (In Japanese).

A method to remove heat generated from the molten fuel, even if a core is molten and extends through a reactor vessel is described. Even if the reactor vessel is failed so that fuel fell, the thus fallen fuel is received in core catchers disposed below the reactor vessel. The fuel fallen into the core catchers produces decay heat but said heat is transmitted to coolant within a coolant vessel disposed externally of the core catchers. This coolant is introduced into a heat exchanger disposed externally of the reactor chamber with heat removed, after which it is pressurized by a circulating pump and returned to the coolant vessel. A low temperature coolant entered through a coolant inlet nozzle passes through a gap between the coolant vessel and the core catchers and flows into the core catchers and is moved out of a coolant outlet nozzle carrying the decay heat within the core catchers. The abovementioned operation is repeated for cooling.

5215 PROCEDURE AND EQUIPMENT FOR REGENERATING COOLING TRAPS IN LIQUID METAL CIRCUITS.

McKee, J.M. Jr. (to Energy Research and Development Administration, Washington, D.C. (USA)). French Patent 2,312,276/A/. 21 May 1976. Priority date 29 May 1975, United States of America (USA), 16p. (In French).

This invention refers to cold traps employed in a liquid metal system to precipitate and remove impurities. It particularly applies to sodium or sodium-potassium systems used for

transferring heat in nuclear energy generating plants. A process was sought that would determine if the cold trap contained an excess of hydride or oxide type impurities and to remove such impurities from the cold traps without having to dismantle them. An improved apparatus has been developed that includes a cold trap and enables the impurities contained in the cold trap to be removed while maintaining the latter in the liquid metal system.

5216 SODIUM LEAK DETECTOR. Iwasaki, T.; Koike, H.; Suzuki, M. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-140,795/A/. 19 May 1976. 3p. (In Japanese).

The object of the invention is to prevent a trouble on the side of a heater from being carried into the sodium loop side and to securely detect leak of sodium. In sodium piping in sodium cooling fast breeder plant, the sodium piping and a heater for preventing sodium condensation are locked through insulations, between which a tray for leaked sodium and a sensor for detection of sodium leakage are mounted.

5217 METHOD OF DETECTING A FUEL ELEMENT FAILURE. Cohen, Paul. (to Energy Research and Development Administration, Washington, D.C. (USA)). Canadian Patent 989,525/A/. 18 May 1976. 10p.

The coolant of a sodium-cooled fast breeder reactor is monitored for the presence of radioactive iodine by equilibrating a sample of the coolant with a molten salt consisting of a mixture of iodides such as the eutectic mixture of barium iodide and strontium iodide, whereby any radioactive iodine ions present in the coolant exchange with iodine ions present in the salt, separating the molten salt from the sodium and monitoring the molten salt for the presence of iodine. If necessary, the molten salt is equilibrated with non-radioactive sodium before monitoring it for the presence of iodine.

5218 FUEL ASSEMBLY. Kato, Y. (to Hitachi Ltd.). Japanese Patent 1977-137,588/A/. 14 May 1976. 3p. (In Japanese).

The object of the invention is to prevent supports of the fuel assembly of a high-flux reactor from becoming fragile due to irradiation with fast neutrons by the provision of a neutron shield between the endless nozzle arrangement of a wrapper tube and fuel pins of the fuel assembly. A neutron shield consisting of stainless steel is provided between an endless nozzle arrangement on the outer periphery of a wrapper tube and the fuel pins of the fuel assembly. Since neutrons passing through the gap in the fuel assembly can be prevented by the shield, it is possible to alleviate excessive stress in the entrance nozzle portion produced due to warping of the fuel assembly at the time of reactor operation.

5219 ROTARY PLUG SEAL DEVICE FOR LIQUID METAL COOLED REACTOR. Sosa, Y. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-135,991/A/. 8 May 1976. 5p. (In Japanese).

The object of the invention is to prevent of metal vapor intruding into the gap between the reactor vessel and shield plug from depositing on the inner side, thereby assuring normal rotary function of the shield plug. A special metal vapor shield mechanism is provided in the gap formed between the reactor vessel and shield plug. This mechanism comprises a

partition member and a gutter accommodating liquid sodium. With this arrangement, even if a great quantity of sodium vapor present in the space above the reactor vessel enters the gap, it can't further proceed upwards by virtue of the shield mechanism. Thus, metal vapor intruding into the gap between the reactor vessel and shield plug can be prevented from depositing on the inner side to thereby guarantee normal rotary function of the shield plug.

- 5220 HEAT EXCHANGERS. Berniolles, J.M.; Jogand, P. (to CEA, 75 - Paris (France)). French Patent 2,350,566/A/. 4 May 1976. 7p. (In French).

This invention relates to a heat exchanger, more particularly intended, although not exclusively, for transferring the thermal power acquired by an initial fluid medium cooling the core assemblies of a nuclear reactor to another fluid medium isolated from the first, therefore having low activity and suitable them to generate steam in a separate generator. The invention applies particularly in the case where the first fluid medium or primary fluid is a liquid metal, generally sodium, used as coolant in a fast nuclear reactor, the second fluid medium or secondary fluid also being liquid sodium, flowing through tubes between two tube plates respectively to enter an intake manifold and an outlet manifold of the secondary sodium. These tubes are externally steeped in the primary sodium. The exchanger concerned comprises two tube plates fitted inside an external envelope crossed by a primary fluid. These two tube plates are joined by a bundle of tubes extending vertically over the major part of their length and through which a secondary fluid flows, these tubes being covered externally by the primary fluid exchanging heat with the secondary fluid through these tubes.

- 5221 SAFETY DEVICE IN A NUCLEAR REACTOR FUEL ROD WITH A CLOSED FISSION GAS PLENUM. Gast, K. (to Gesellschaft fuer Kernforschung m.b.H., Karlsruhe (Germany, F.R.)). German(FRG) Patent 2,023,250/B/. 29 Apr 1976. 2p. (In German).

The invention relates to the design of the fission gas plenum in fuel rods of liquid metal cooled nuclear reactors. The fission gas plenum is subdivided by separation walls (preferably made of sinter metals) arranged in a honeycomb structure, which are fully permeable to fission gases. In this way a gradual, uniform pressure buildup is achieved, and in case the fuel rod cladding tube were to burst in this region, the amount of fission gas accumulated would be released only after considerable delay. The safety of such fuel elements is enhanced by the invention.

- 5222 LIQUID METAL/WATER HEAT EXCHANGER WITH REPLACEABLE TUBE BUNDLES. Jansing, W.; Vinzens, K. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). German(FRG) Patent 2,448,832/A/. 22 Apr 1976. 11p. (In German).

The invention relates to an improvement of liquid metal-water exchangers used in nuclear power plants with the purpose of reducing the time needed to replace faulty tubes or components and to repair defects. For this purpose it is suggested to arrange the vertical tube bundles, each of which is surrounded by a shell open at the top and the bottom, so as to make them and the shell removable from the top and to close the free cross section between the

inner vessel wall and the outer shell wall. Other advantageous designs are indicated, above all those protecting the cold feedwater tubes from the hot metal.

- 5223 METHOD AND DEVICE FOR THE DETECTION OF A FAILED FUEL ELEMENT. Fryer, R.M.; Matlock, R.G. (to Energy Research and Development Administration, Washington, D.C. (USA)). German(FRG) Patent 2,546,034/A/. 22 Apr 1976. 9p. (In German).

The detection of the failure of a fuel element in a metal-cooled fast breeder reactor makes difficulties. It is proposed to use the fission gas released thereby as indicator. By the example of the EBR-II nuclear reactor it is described how each fuel subassembly is provided at the outlet with a gas trap - shaped like a reversed funnel with a valve - which may be individually read in the known manner so that a leak may be found without great effort.

- 5224 SEE-THROUGH DEVICE FOR A REFUELING MACHINE. Ito, J.; Sato, T. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-127,590/A/. 20 Apr 1976. 4p. (In Japanese).

A method to make a fuel assembly visible without moving a grip of a refueling machine to enhance a locating accuracy of the refueling machine is described. In a see-through device, a pantagraph is opened before a rotary plug is rotated, and a standucer disposed at the foremost end of the pantagraph is inserted into a lower guide tube of a fuel assembly through a side hole. This transducer introduced into the lower guide tube has transmitting and receiving surfaces directed downwardly. In this state, the rotary plug is rotated to move the refueling machine to a desired position. The transducer transmits and receives ultrasonic waves until the refueling machine has been located accurately. After the refueling machine has been located, the pantagraph is closed and the transducer is removed from the lower guide tube and the fuel assembly is gripped by a gripper of the refueling machine.

- 5225 LIQUID METAL COOLING REACTOR. Ikeda, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-125,996/A/. 15 Apr 1976. 6p. (In Japanese).

A guide passage which spirally swivels a coolant, within a coolant inlet nozzle to sufficiently mix the coolant flowing into the lower plenum is described. A coolant inlet nozzle is mounted at the bottom of a reactor vessel at a right angle to the wall surface of the reactor, and two wing plates radially twisted with each other are disposed within the inlet nozzle. As the coolant passes through the wing plates, these wing plates serve as a guide passage to create a swivelling flow whereby the coolant flows into the lower plenum in the form of a swivelling flow and is dispersed in a radial direction by a centrifugal force to be mixed with another coolant already present therein.

- 5226 HEAT EXCHANGER PROVIDED WITH PLATES. Chaix, J.E.; Fajean, M.; Chlique, B. (to CEA, 75 - Paris (France); Fives-Cail-Babcock, 75 - Paris (France)). French Patent 2,348,458/A/. 12 Apr 1976. 13p. (In French).

The invention relates to a heat exchanger of the plate type, in which two fluids exchange calories through parallel metal plates, delimiting spaces separated from each other in which two fluids respectively flow without

direct contact between them. The invention particularly applies in the case where one of the two fluids is water under pressure or else a circulating liquid metal, specially sodium, used in the system of a pressurised water or fast neutron reactor, the second fluid being water to be vaporised in the exchanger by the calories supplied by the first fluid. The arrangement is designed to give minimum bulk, particularly enabling the exchanger to be housed in the area between the core of a nuclear reactor and a casing or outer vessel, or else in an external sealed containment, with a view to recovering with the best efficiency the heat acquired by a coolant flowing through the core.

- 5227 MAINTENANCE CASK. Koizumi, K.; Yoshizawa, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-122,795/A/. 9 Apr 1976. 3p. (In Japanese).

A maintenance cask to be used for a fast breeder as unit assembly comprising a sealing structure and a shield structure separate from each other is described. This maintenance cask comprises a door valve, a plurality of cylindrical casks stacked one above another on the door valve, a grid shaft seal member hermetically joined to the upper surface of the cask stack, a door valve shield covering the door valve, a cap member and a gripper. Thus, it is possible to assemble a maintenance cask to meet the shape of the associated apparatus by suitably using appropriate component parts.

- 5228 PROCESS AND DEVICE OF THERMAL SHIELDING OF THE VESSEL OF A NUCLEAR REACTOR. Gamma, J.M.; Paillard, L. (to CEA, 75 - Paris (France)). French Patent 2,347,749/A/. 6 Apr 1976. 10p. (In French).

This invention relates to the thermal protection of the vessel of a liquid metal cooled integrated nuclear reactor. The process consists in circulating along the vessel part of the liquid metal injected at the bottom of the core. In a leak tight annular chamber above the flowing liquid metal, an inert gas (argon) is placed under pressure to regulate the gas-liquid metal interface in this chamber so as to obtain a distribution of temperatures along the vessel so that the range of stress changes is restricted. According to a first variant of the invention, the inert gas pressure is created by trapping in the chamber a mass of argon under an initial pressure so that the level of the gas-liquid metal interface in this chamber is kept appreciably constant during the power changes of the reactor. According to a second variant of the invention, this pressure is achieved by trapping in the chamber a mass of argon, the pressure of which is adjusted, whilst the reactor is working, so that the power changes of the reactor generate movements of the gas-liquid metal interface that will maintain the thermal gradient, created along the vessel, at an appreciably constant and sufficiently low level.

- 5229 SHIELDING PLUG DEVICE. Orif, S.; Hasegawa, S.; Makishima, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-121,196/A/. 5 Apr 1976. 3p. (In Japanese).

A method to reduce the size of and extend the life of a revolving bearing and facilitate the laying of driving cables and duct lines by providing plug raising means of a fast breeder on a stationary plug mounting base so as to prevent the shearing force of sodium from acting upon the revolving bearing is described.

The shield plug means comprises a stationary plug secured to the open end of the reactor container, a rotary plug rotatable with respect to the stationary plug, an annular base formed on top of the stationary plug so as to cover the rotary plug, a bearing secured to the rotary plug edge lower face and upper and lower locking plates. At the time of the rotation of the rotary plug, the upper locking plate is withdrawn, the stationary plug is raised to release the seal structure, and the lower locking plate is inserted between the bearing and stationary plug. In this way, smooth rotation of the rotary plug can be obtained.

- 5230 NUCLEAR REACTOR. Wade, E.E. (to Westinghouse Electric Corp., Pittsburgh, Pa. (USA)). French Patent 2,307,339/A/. 2 Apr 1976. Priority date 9 Apr 1978, United States of America (USA), 15p. (In French). Priority claim: 9 Apr 1975, US.

A nuclear reactor is described which includes a cylindrical pressure vessel, a nuclear core formed of an assembly of fuel elements held in the pressure vessel, a structure for lowering the nuclear reactor core, placed above the core to maintain the fuel elements in position, and at least two rotating plugs maintained in the upper wall of the receptacle, one of the plugs being fitted excentrically in relation to the other plug. It includes fuel element handling gear and is characterised by the lowering structure being composed of at least two parts corresponding to the rotating plugs and each part of the lowering structure is connected to the corresponding rotating plug to move with it.

- 5231 REACTOR CONTROL ROD. Saijo, T. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-121,198/A/. 2 Apr 1976. 4p. (In Japanese).

A method is described to enable quick descent of a control rod body even when some relative phase deviation between upper drive means and wrapper tube is produced, while permitting a coolant to effectively flow into a protective tube irrespective of the position of the control rod body. In a control rod used for a nuclear reactor such as a fast breeder, an orifice which dispenses with a cylindrical guide tube and has a greater inner diameter than the outer diameter of the protective tube of the control rod body is provided on the inner side of a wrapper tube, thus permitting smooth operation of the control rod body and also permitting the coolant to effectively flow into the protective tube irrespective of the control rod body.

- 5232 THERMAL PLANT FOR ENERGY PRODUCTION. Tillequin, Jean. (to Fives-Cail-Babcock, 75 - Paris (France)). French Patent 2,346,815/A/. 31 Mar 1976. 11p. (In French).

This invention relates to an energy production plant to serve a variable demand network, the basic thermal source of which is composed of a nuclear reactor, particularly of the fast neutron type and sodium cooled. The excess power produced during off-peak hours is stored in a heat accumulator for subsequent return to the network during peak periods. The heat accumulator is formed of a tank filled with a thermal fluid, a charging circuit is fitted to take heat on demand from the reactor, store it in the accumulator and a return circuit is fitted to release on demand the heat stored in the accumulator. The charging circuit is fitted to take it from the liquid sodium flowing through a cooling loop of the reactor and the return circuit is fitted to release

this heat to a steam generator that feeds an auxiliary steam turbine. Under this invention, the charging circuit is connected to the upper part of the accumulator so as to feed liquid sodium to it; this liquid sodium thus forming the thermal fluid for filling the tank, and to the bottom of the accumulator so as to return the liquid sodium to the cooling loop.

- 5233 FAST REACTOR. Berniolles, J.M.; Jogand, P. (to CEA, 75 - Paris (France)). French Patent 2,346,816/A/. 29 Mar 1976. 13p. (In French).

This invention relates to a fast reactor, using a novel arrangement for the heat exchangers fitted between the main vessel and the internal cylindrical vessel. The particular purpose of this lay-out is to ensure a better distribution of the hot liquid metal at the intake of the exchangers and better thermal insulation between the hot liquid metal and the cold liquid metal taken from the space separating the main vessel from the internal vessel. The invention also aims to allow a significant simplification in the design of the connection sleeves between the exchangers and the internal vessel.

- 5234 FAST BREEDER. Miki, K.; Inoue, K. (to Hitachi Ltd.). Japanese Patent 1977-115,991/A/. 26 Mar 1976. 3p. (In Japanese).

The object of the invention is to disperse curved directions of fuel assemblies due to the difference in temperature of walls of a wrapper tube and making use of interaction of adjacent fuel assemblies to limit deformation, thereby minimizing the deformation of a core. Two kinds of fuel assemblies in which wire spacers wound about fuel pins are opposite in rotative direction with each other are alternately arranged in each layer to constitute a core. Therefore, the fuel assemblies arranged in the radial direction of the core are different in curved direction in each layer, and the amount of deforming the core in the radial direction is restricted by the adjacent fuel assemblies. As a consequence, a load imposed on the fuel assembly near the center of the core due to a compulsive reaction from a restriction mechanism around the core is less than the conventional core. It will be noted that the curved direction of each fuel assembly is determined depending on pitch of the wire spacer.

- 5235 LIQUID-METAL COOLING REACTOR. Mitsutsuka, N.; Kubo, K.; Miyazawa, T. (to Tokyo Shibaura Electric Co. Ltd.). Japanese Patent 1977-115,992/A/. 25 Mar 1976. 9p. (In Japanese).

The object of the invention is to quickly and positively detect a failure condition of nuclear fuel inside the reactor to enhance safety. A coolant in contact with a core partly passes through a sampling pipe, a solenoid pump, a flow meter, a tank and a pipe. When a nuclear fuel charged in a core is failed, a nuclear fission formation gas in the coolant moves into a space above said tank. This gas flows into a first and second nuclear fission formation gas detectors, and a signal is sent to a system of failed fuel detection and diagnosis. The coolant discharged from the tank flows into ^{135}I detection system and is returned to a reactor vessel. At this time, an adsorption tank is adjusted in temperature, and after a lapse of given time, the flowing of coolant into the ^{135}I detection system is stopped for detection of $\text{sup}(^{135}\text{m})\text{Xe}$ and ^{135}Xe formed from ^{135}I by $\text{sup}(^{135}\text{m})\text{Xe}$ detection

portion and output signals from said detection portion are processed to calculate the amounts of $\text{sup}(^{135}\text{m})\text{Xe}$ and ^{135}Xe , and the amount of ^{135}I in the coolant from the operation conditions of the ^{135}I detection system.

- 5236 EMERGENCY SYSTEM FOR NUCLEAR REACTORS. (to CEA, 75 - Paris (France)). French Patent 2,345,792/A/. 24 Mar 1976. 8p. (In French).

The invention concerns a circuit called 'of emergency help' intended to remove, in a safe and quick manner, the residual thermal power on the safety vessel of a fast neutron reactor cooled by a liquid metal flow, in the event of a failure occurring inside the main reactor vessel or on it. This system includes a network of spray nozzle tubes, distributed around and near the external surface of the safety vessel, to project on to the surface of the vessel a mist of a liquid having high latent vaporisation heat. The steam produced on contact with the safety vessel is collected in the space provided between the safety vessel and the external protection vessel by at least one collector pipe for discharging this steam outside the vessel. Under a preferred design mode of the invention the liquid is water the use of which turns out to be particularly advantageous in practice owing to its favourable physical properties and its low cost.

- 5237 FUSE APPLIED TO DESIGNING A SCRAM DEVICE FOR NUCLEAR REACTOR. Taulier, H.H.L. (to Electricite de France, 75 - Paris). French Patent 2,345,709/E/. 24 Mar 1976. 11p. (In French).

This invention concerns a fuse and its application to the building of an emergency shutdown system for liquid sodium cooled fast reactors. The object of this system is to shut down the reactor automatically without manual intervention when an operating anomaly might cause the core fuel elements to melt. There are already aluminium fuses making up a charge in an absorbant substance forming one piece with a fixed part of the reactor and suspended above the free channel of the reactor core. In normal operation, the aluminium fuse melts at 800°C . At normal reactor operating temperature (around 560°) it is found that the fuse creeps and this could prematurely trigger off the emergency shutdown system. The object of this addendum is to improve the fuses described in the main patent specification, making it possible to achieve the cooling of the emergency shutdown system and therefore of the fuse by the sodium coolant of the reactor, whilst retaining the essential characteristics of the fuse. This fuse is not affected by any creep that might occur and this prevents an undesirable triggering off of the emergency shutdown system.

- 5238 LOW ALLOY STEEL FOR A HEATING PIPE OF A FAST BREEDER STEAM GENERATOR. Moroishi, T.; Shida, Y.; Yukitoshi, T.; Yoshikawa, K. (to Sumitomo Metal Industries Ltd., Osaka (Japan)). Japanese Patent 1977-110,398/A/. 12 Mar 1976. 6p. (In Japanese).

Object: To provide a low alloy steel for high temperature use, having excellent high temperature strength, tenacity, weldability and very small decarbonization under conditions of liquid sodium at a high temperature. Structure: A low Cr-Mo alloy steel having a structure which consists of from 0.02 to 0.15% C, from 0.10 to 1.50% Si, from 0.10 to 1.50% Mn, from 1.00 to 8.00% Cr, from 0.30 to 1.50% Mo, and balance essentially Fe. The structure is at

least one to three mixture out of ferrite, pearlite, tempered bainite or tempered martensite, and has untempered structure of either martensite or bainite.

- 5239 FUEL CAN FOR A LIQUID-METAL COOLED FAST BREEDER. Ozaki, S. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-110,396/A/. 12 Mar 1976. 3p. (In Japanese).

A method to prevent effusion and migration of radioactive Co-60 produced by irradiation with neutrons into a coolant, and to reduce a concentration of the Co-60 in the coolant is described. A nickel alloy containing not more than 0.1% cobalt is coated on a surface of a fuel cladding tube of an alloy containing cobalt as impurity. The nickel alloy is a stainless steel, a nickel steel, a nickel base alloy, etc. Thickness of the cladding layer may be an order enough to withstand corrosion and wear during service (Generally 25 microns or more).

- 5240 FAILED FUEL DETECTOR. Takahara, K. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-106,095/A/. 3 Mar 1976. 5p. (In Japanese).

A method to eliminate a driving means such as a selector valve in a failed fuel detector of a sodium cooled fast reactor, by sampling a coolant with an ejector is described. There are provided a coolant-receiving opening facing to a coolant outlet of each fuel assembly, an ejector connected to each receiving opening, a gas-feeding mechanism for feeding a given gas to these ejectors, a container containing the coolant from a piping connected to a discharge side of each ejector, and a detector for detecting nuclear fission products in the container. The failed fuel detector is so arranged as to take a sample of the coolant in the fuel assembly in the container by means of an ascending force in the coolant of gas fed to the ejectors.

- 5241 FAST BREEDER REACTOR AND FUEL ASSEMBLIES FORMING THE CORE. Lunt, A.R. (to UKAEA London Office). French Patent 2,303,351/A/. 2 Mar 1976. Priority date 3 Mar 1975, United Kingdom of Great Britain and Northern Ireland (UK), 8p. (In French).

A description is given of a fast breeder reactor cooled by a liquid metal and comprising a core formed of several long fuel assemblies rising above the core bearing system. The feature of this reactor is that each assembly includes an inertia damper system in the region of the free upper end.

- 5242 FAILED FUEL DETECTOR IN REACTOR. Mitsuoka, N.; Miyazawa, T.; Kubo, K. (to Tokyo Shibaura Electric Co. Ltd.). Japanese Patent 1977-104,691/A/. 27 Feb 1976. 7p. (In Japanese).

The object of the invention is not only to make the detector small, but also to increase the detection sensitivity for concentration of nuclear fission iodine. Structure: Liquid sodium, a coolant, is introduced into a gas-liquid separator through a piping at an upper portion of a reactor container, and then into an adsorption container. At this time, Xe-135m produced by disintegrating I-135 in the liquid sodium migrates to a cover gas at an upper portion of the gas-liquid separator. I-135 introduced into an adsorber is cooled to 200°C or less in a cooler and is adsorbed on an adsorbent. Flow of the liquid sodium is stopped and Xe-135m in the adsorbent is migrated into

the cover gas by heating the adsorber to 300°C or more, as a determined quantity of the liquid sodium is extracted from the reactor container. Therefore, the cover gas is introduced into a radioactivity measuring instrument.

- 5243 NUCLEAR FUEL ELEMENT. Shimada, M. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-101,384/A/. 20 Feb 1976. 2p. (In Japanese).

The object is to impart a sodium-resisting property to a cladding tube for preventing selective effusion of the alloy constituents in the liquid sodium, thereby preventing creation of deterioration in mechanical properties of cladding tube. In a nuclear fuel element, wherein a nuclear fuel material is received into a cladding tube, opposite ends of which are sealed by end plugs, the inner surfaces of the cladding tube comprise a layer of material which produces no swelling by irradiation of fast neutron, whereas the outer surfaces thereof comprise a layer not corroded by liquid metal.

- 5244 METALLIC COMPONENT WITH A CHROMIUM CARBIDE BASE PROTECTIVE COATING. Wolf, T.A.; Tucker, R.C. Jr. (to Union Carbide Corp., New York (USA)). French Patent 2,300,139/A/. 9 Feb 1976. Priority date 10 Feb 1975, United States of America (USA), 10p. (In French). Priority claim: 10 Feb 1975, US.

The invention concerns a coating system to protect metal components in sodium and helium cooled nuclear reactors. It includes a nickel or iron based alloy metal substrate, a first coat formed on the substrate and comprising chromium carbides and a binder selected among the chromium-nickel, chromium-cobalt, chromium-iron alloys and the super-alloys, the first coating being between 25 and 380 microns thick, and a surface coating comprising pure chromium carbides and being between 12.5 and 125 microns thick.

- 5245 ABSORBING SEALS FOR NUCLEAR REACTORS. Cooper, M.H. (to Westinghouse Electric Corp., Pittsburgh, Pa. (USA)). French Patent 2,301,072/A/. 5 Feb 1976. Priority date 13 Feb 1975, United States of America (USA), 13p. (In French).

Priority claim: 13 Feb 1975, US.

The system described is for preventing the leakage of short half life radioactive isotopes, from a gas contained in nuclear reactor vessels and crossing the various junctions of such vessels. This system includes seals located in the junctions and its feature is the system of retentions to contain all the isotope leaks passing through the seals during a sufficient time to allow these isotopes to decay until they reach a harmless concentration before they are released from this retention system.

- 5246 FAST BREEDER OF LOOP TYPE LIQUID METAL COOLING SYSTEM. Nakai, Y.; Hoshi, Y. (to Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Japanese Patent 1977-92,090/A/. 28 Jan 1976. 3p. (In Japanese).

The object of the invention is to locate an upper edge of a guard vessel of each apparatus in a primary liquid metal system lower than a system level to shorten an axial length of a primary circulating pump and to form each guard vessel into a smaller size. Upper portions of guard vessels for a reactor vessel, an intermediate heat exchanger and a primary

circulating pump are disposed in position other than those in the vicinity of piping which forms a riser and are closed at a height lower than a system level, whereby a difference between the system level and an emergency level is reduced to shorten an axial length of a primary circulating pump.

- 5247 DETECTION SYSTEM OF SODIUM OXIDE VAPORS. Hundal, R. (to Westinghouse Electric Corp., Pittsburgh, Pa. (USA)). French Patent 2,299,640/A/. 26 Jan 1976. 6p. (In French).

The description is given of a sodium oxide vapor detection system which comprises a containment, a light source located to send a light beam into and through this containment and a photodetector located to intercept the light from the source after it has covered a given path through the containment. In response to the intensity of the incident light, the photodetector produces an output signal. The feature of this device is a first polarizer located near the light source, along the path of the light coming from it and designed to polarize the light projected through the containment in a given plane, and a second polarizer located near the photodetector, along the path of the polarized light and designed virtually to prevent all the light rays whose orientation differs from the given polarization plane from reaching the photodetector.

- 5248 FUEL ASSEMBLY. Uesugi, N. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-89,795/A/. 23 Jan 1976. 4p. (In Japanese).

The object of the invention is to put a mark that may detect displacement in position between fuel assemblies to reduce error in signal. The device comprises a bundle of a plurality of fuel elements, a shroud which surrounds the outer peripheral surface of said bundle, a handling head positioned on said shroud, and discriminating marks and checking marks impressed on the upper end surface of said handling head, said discriminating marks having a single mark impressed to indicate a reference at the starting point thereof.

- 5249 DEVICE FOR DETECTING RADIOACTIVITY IN A LIQUID METAL COOLANT. Miyazawa, T.; Kubo, K.; Taguchi, J. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-89,796/A/. 22 Jan 1976. 4p. (In Japanese).

The object of the invention is to considerably increase efficiency of neutron detection and at the same time, to facilitate maintenance and checking and to increase safety. A neutron detection system is provided in the outer periphery of a dual pipe line which extends upwardly from a cylinder type induction system magnet pump so that when a fissionable product is contained in the flowing coolant, a neutron released from said fissionable product is detected by the neutron detection system. In this case, the neutron detection system is mounted so as to surround the dual pipe line portion and can detect most of neutrons released over the range of 360 degrees. Since the neutron detection system and a coil in the cylinder type induction magnet pump are insertably mounted in the outer periphery of the dual pipe line portion wherein the coolant is flown from an outer passage towards an inner passage in a part of a piping line through which coolant flows, they can be merely pulled out and new ones may be fitted, in case of replacement.

- 5250 FUEL ASSEMBLY. Takeda, S. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-89,794/A/. 21 Jan 1976. 5p. (In Japanese).

The object of the invention is to make a distribution of coolant flow in a radial direction and an azimuth direction within a core of a fuel assembly as flat as possible as the fuel burns. An orifice formed in a side of an entrance nozzle at the end of a shroud is designed to have an area which varies in the azimuth direction to decrease the coolant flow and capacity of pump.

- 5251 CONTROL ROD FOR A FAST BREEDER. Saito, T.; Yamagishi, Y.; Koga, T. (to Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Japanese Patent 1977-87,598/A/. 16 Jan 1976. 3p. (In Japanese).

The object of the invention is to dispose a stopper, which is destroyed when a reactor power increases, at an upper part within a shroud and to place a neutron absorber on the stopper so as to constitute a reactor stopping system which is simple in construction and high in accuracy. When the reactor power increases as a result of failure, fissionable material or a stopper whose melting point is the limit temperature at the core is destroyed and a neutron absorber moves down within the shroud until it comes into contact with a cushion material by means of a spring for the scrambling operation.

- 5252 METHOD OF CONTROLLING THE SECONDARY COOLANT IN FBR. Kajiyama, S.; Nakamura, A.; Watanabe, T. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-87,599/A/. 14 Jan 1976. 5p. (In Japanese).

The object of the invention is to stabilize and control flow rate without mutual interference by providing a closed-circuit flow control system for controlling the flow of secondary coolant to a superheater or a reheater, and a flow control system having an operator in which a relationship between the flow, opening degree of an adjusting valve and pressure is predetermined. A secondary coolant flow-signal detected by a detector and supplied from an intermediate heat exchanger to a first pipe line is fed into a ratio setting instrument, a second set signal obtained by a ratio of a first set signal is fed into an adjusting meter, a first adjusting valve is opened and closed on the basis of difference from the detection signal of the detector, the flow of secondary coolant branched into a second pipe line and supplied to the superheater is controlled in constant, a third set signal from the ratio setting instrument is fed into the operator to obtain a degree of valve opening, and a second adjusting valve is opened and closed to control in constant the flow of secondary coolant branched to a third pipe line and supplied to the reheater.

- 5253 LOW-SWELLING STAINLESS KEY TO LMFBR DEVELOPMENT. Obrzut, J. Iron Age; 217: No. 2, 34-35 (12 Jan 1976).

Stainless steel Type 316 or any other stainless used in liquid-metal fast breeder reactors can swell deleteriously under certain conditions by bombardment with intense high-energy neutrons. This irradiation-induced swelling can be reduced 5 to 10 times by additions of Ti and Si to the stainless composition. A table of Type 316 and alloys with lower Cr compositions with Ti and Si additions is shown. With such modified chemistry more Pu can be bred from each fission

atom and the breeder reactor will be able to operate at higher power levels. A test was developed to screen alloys quickly by bombarding the test alloys with Ni ions from the Oak Ridge National Laboratory's 5.5 MV Van de Graaff accelerator. Cold working could possibly further improve the resistance of the new alloys to swelling.

- 5254 TESTING EXPERIMENTAL FUEL ELEMENTS OF THE BN-600 FUEL ELEMENT TYPE UP TO VARIOUS DEPTH OF BURN UP IN THE BOR-60 REACTOR. Antipina, M.M.; Bibilashvili, Y.K.; Golovin, I.S. At. Energ. (USSR); 40: No. 1, 16-27 (Jan 1976). (In Russian).

Results of the investigation of experimental fuel elements are presented. The authors discuss fuel element construction, basic testing parameters, results of measuring gas release from fuel, deformation of cladding and swelling of steel, and also data on material investigations of macro- and micro-structures of fuel and cladding with an analysis of the degree and character of their physico-chemical interaction with fission fragments.

- 5255 NUCLEAR PROSPECTS. THE AMERICAN VIEW. S. Afr. J. Sci.; 72: No. 1, 3-4 (Jan 1976). BREEDER REACTORS; CANDU TYPE REACTORS; LMFBR TYPE REACTORS; LWBR TYPE REACTORS; NUCLEAR INDUSTRY; NUCLEAR POWER; PLANNING; USA; WASTE MANAGEMENT; WATER MODERATED REACTORS

- 5256 TRANSIENT ANALYSIS OF THE PRIMARY SYSTEM OF A LIQUID METAL-COOLED FAST BREEDER REACTOR PLANT. Pavlenko, G.F. Ithaca, NY; Cornell Univ. (1976). 360p. University Microfilms Order No. 77-8402.

Thesis (Ph. D.).

A computer-aided simulation is developed to analyze the transient behavior of the primary system of liquid-metal-cooled fast-breeder-reactor (LM GAMMA BR) plants of the loop type. A set of nuclear, thermal, hydraulic, and control models for the primary system is developed to simulate the transient response of sodium cooled fast breeder reactors for normal operating conditions, for a class of postulated accidents which do not disturb the system geometry, and for a major pipe rupture accident. The analytical submodels of the primary system components and systems are assembled into five major models which are coded in the FORTRAN IV language to form five computer codes. These major models simulate: (a) the dynamics of sodium flows in the primary system governed by forced and natural circulation effects and the operation of the primary pumps and primary flow/speed control system, (b) reactor heat-transfer, (c) neutron kinetics, reactivity feedbacks, and reactor power control, (d) intermediate heat exchanger heat-transfer, and (e) sodium transport in the primary piping system. The primary system code is used to analyze five representative transients: (a) ramp change of power level, (b) single pump coastdown, (c) reactor scram shutdown, (d) coastdown of all primary pumps, and (e) loss of coolant accident. The code can be used to analyze other transients as well, of similar or of different types, using appropriate sets of input data or by making minor modifications which can be easily accommodated.

- 5257 SPACE--TIME DYNAMICS OF A FAST BREEDER REACTOR FOR LOCALIZED DISTURBANCES. Doshi, J.B. Berkeley, CA; Univ. of California (1976). 238p. University Microfilms Order No. 77-4439.

Thesis (Ph. D.).

Sodium voiding accidents in Liquid Metal Fast Breeder reactors may cause a positive reactivity insertion due to a lack of balance between two competing effects following the voiding: (1) the shift in neutron energies to higher values which leads to a positive reactivity addition, and (2) the increased neutron diffusion which generates a negative reactivity. Since these two effects have strong spatial dependence, the amount of reactivity added in a voiding accident depends strongly on the position in the reactor core where the voiding accident has occurred. A method of analysis is developed for the localized voiding perturbation based on a flux factorization technique, accounting for the flux shape changes taking place near the region of the voided subassembly. Heat transfer equations are solved in conjunction with the neutronic equations and the two are coupled via feedback effects like the Doppler effect. For the purpose of analysis, a typical large (1000 MWe) LMFBR with two radial core-zones of different enrichments is considered. Time-dependent void fractions, which are the input quantities in the analysis, are taken to be typical of a channel flow blockage accident. The transients are analyzed for voidings taking place at three different radial locations in the core and for two different rates of voiding. Results of the calculations show that the voiding transients occurring at the center of the reactor core cause most severe effects in terms of the positive reactivity addition and the effects become progressively less severe for voidings occurring outward toward the periphery of the reactor.

- 5258 PARAMETRIC APPROACH TO LMFBR SAFETY EVALUATION. Burns, R.D. III. Lafayette, IN; Purdue Univ. (1976). 304p. University Microfilms Order No. 77-7424.

Thesis (Ph. D.).

The assessment of risk from LMFBRs has evolved to the present "design basis accident" approach. Of particular interest are consequences from low probability "hypothetical" accidents. Due to uncertainties involved in the accident progression, accident consequence spectra are needed to represent the probabilistic nature of the results of accident simulations. Many single simulations are needed in order to create probabilistic distributions of consequences. For these reasons, the accident simulation code PARSEC was developed. It runs more than two orders of magnitude faster than existing codes and is designed for use in parametric studies of uncertain input values. Parametric sensitivity studies show the relative impact of uncertainties on the results of the accident simulation. They indicate particular sensitivity to reactivity coefficients and core heat capacity. Many parameters, especially those associated with hydraulics calculations, show little sensitivity. An example of using simulation results for calculation of probabilistic accident consequence spectra shows wide variability of simulation results. The distributions appear to be largely symmetric, due to the lack of treatment of any highly nonlinear effects and the use of symmetric distributions for probabilistic input. The results show that the probability of realizing "limiting" or "bounding" case consequences is far out on the tail of the distribution, and are of vanishingly small probability. A comparison of PARSEC and SAS3A results for a 5 channel, unprotected loss-of-flow (LOF) accident shows that PARSEC predicts the core's behavior well. By applying a probabilistic

treatment of uncertainties in six input values, the errors in the PARSEC models are shown to be small compared to the overall uncertainties in the results themselves.

- 5259 PROSPECT OF FAST BREEDER REACTOR DEVELOPMENT. Ichino, I. (Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). pp 338-360 of Variety of problems in promotion of nuclear power generation. Tokyo, Japan; Fuji International (1976). (In Japanese)

Fast breeder reactors (FBR) are required because of the effective use of energy source or nuclear fuel. Besides, the influence on the environment is small, and the improvement of economy can be anticipated according to the future development toward low pressure and small size. In Japan, the diversification of energy sources is required to maintain the energy consumption and also to correct the excessive dependence on oil. From this viewpoint, the development of nuclear power generation, especially of FBRs, is desirable more than the significance of the national project. It is supposed that the development schedule for breeders must be restudied sooner or later. Though breeders have problems to be solved for the practical use, such as fuel, sodium technology and others, powerful investigation is being carried out in several countries because they hold the key for the solution of energy problems. The status of the development in the U.S., U.K., France and German Federal Republic is surveyed. In Japan, the development seems to be progressing with the steady procedure to aim at the practical use in 1980s through the step of experimental, prototype and demonstration reactors, though the start was late. The projects for the experimental fast reactor "Joyo" and the prototype "Monju" are in progress and the later is being prepared for the start of construction in 1977. Many experimental and developmental tests have been performed in parallel with the above projects. The related testing facilities are described.

- 5260 EXPERIENCE IN WELDING OF THE WEATHER-PROOF STEEL OUTER SHELL OF A REACTOR CONTAINMENT. Herberg, G. (Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). Schweissen Schneiden; 28: No. 6, 227-228(1976). (In German).

Measures for quality assurance in welding the outer shell of WSt 52-3 steel of the SNR 300 nuclear power plant in Kalkar/Niederrhein are presented, the causes for the microcracks found in welded joints (light microscope, microprobe) are also investigated. These occurred above all when using stick electrodes with alloyed cladding, less with electrodes alloyed in the core and hardly in CO₂ weldings. These can be practically avoided by limiting the copper content in the welding material to about 0.40%.

- 5261 RADIOLOGICAL SIGNIFICANCE OF TRANSURANIUM RADIOISOTOPES RELEASED TO THE ENVIRONMENT DURING OPERATION OF THE LMFBR FUEL CYCLE. Barr, N.F. (Energy Research and Development Administration, Washington, D.C. (USA)). pp 649-655 of Transuranium nuclides in the environment. Vienna; IAEA (1976).

From Symposium on transuranium nuclides in the environment; San Francisco, Calif., USA (17 - 21 Nov 1975).

Estimates based on current knowledge and conservative assumptions indicate that release of transuranium elements from the Liquid Metal Fast Breeder Reactor (LMFBR) fuel cycle are

likely to produce population dose commitments small compared to those produced by naturally occurring alpha emitters and globally dispersed transuranium radioisotopes from tests of nuclear weapons in the atmosphere. Potential health consequences of these releases to current and future generations are estimated to be very small compared to risks associated with the production of energy by fossil fuels. The estimates are subject to a number of uncertainties imposed by lack of knowledge. Some of the uncertainties are not likely to be greatly reduced until LMFBR facilities are designed and operated. Others may be significantly reduced prior to facility design and operation. The paper discusses the sensitivity of the estimates to uncertainties and approaches to reducing those uncertainties that strongly influence the estimates.

- 5262 SOME CONSIDERATIONS ON FAST BREEDER REACTOR PENETRATION IN ITALY. Baldo, L.; Viotti, V. (Progettazioni Meccaniche Nucleari, Genoa (Italy)); Lanz, R.; Musso, B. (Nucleare Italiana Reattori Avanzati (NIRA) Genova, Italy). Energ. Nucl. (Milan); 23: No. 8, . 423-236(1976). (In Italian).

This paper deals with the possible role of liquid metal fast breeder reactors (LMFBR) in the Italian electric power system, in an optimized utilisation strategy of natural nuclear fuel resources. The effects on fast breeder reactor penetration of possible alternative policies of light-water and heavy-water nuclear plants installation are studied. Core design and fuel cycle parameters, limiting LMFBR maximum penetration in supplying a large portion of primary energy needs of the country, are also discussed.

- 5263 APPLICATION OF PROBABILISTIC METHODS IN THE FIELD OF NUCLEAR SAFETY. Carrino, A. (CEA Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France). Dept. de Surete Nucleaire). Rev. Gen. Nucl.; No. 5, 406-419(1976). (In French).

Beyond the determination of the risks of accident to the reactors (Rasmussen Report), the probabilistic methods have other fields of application insofar as nuclear safety is concerned. These different fields of application are studied, i.e.: analysis of the accidents and of system performance, research for solutions to obtain an improved reliability, determination of the operating rules of the plants and drawing up of the regulations. In order to illustrate the work tending to improve the reliability, the studies concerning the design of the emergency shutdown system of the Super-Phenix are presented as an example.

- 5264 STATISTICAL OPERATION OF NUCLEAR POWER PLANTS. Gauzit, M. (Alsthom, 75 - Paris (France)); Wilmart, Y. (Novatome Industries, 92 - Le Plessis-Robinson (France)). Rev. Gen. Nucl.; No. 3, 236-248(1976). (In French).

A comparison of the statistical operating results of nuclear power stations as issued in the literature shows that the values given for availability and the load factor often differ considerably from each other. This may be due to different definitions given to these terms or even to a poor translation from one language into another. A critical analysis of these terms as well as the choice of a parameter from which it is possible to have a quantitative idea of the actual quality of the operation obtained is proposed. The second section gives, on an homogenous basis and from the results supplied by 83 nuclear power stations now in

operation, a statistical analysis of their operating results: in particular, the two light water lines, during 1975, as well as the evolution in terms of age, of the units or the starting conditions of the units during their first two operating years. Test values thus obtained are compared also to those taken 'a priori' as hypothesis in some economic studies.

- 5265 **RADIOLOGICAL SIGNIFICANCE OF TRANSURANIUM RADIOISOTOPES RELEASED TO THE ENVIRONMENT DURING OPERATION OF THE LMFBR FUEL CYCLE.** Barr, N.F. (Energy Research and Development Administration, Washington, D.C. (USA)). pp 649-655 of Transuranium nuclides in the environment. Vienna; International Atomic Energy Agency (1976).

From IAEA international symposium on transuranium nuclides in the environment; San Francisco, CA, USA (17 Nov 1975).

See CONF-751105--.

Estimates based on current knowledge and conservative assumptions indicate that release of transuranium elements from the Liquid Metal Fast Breeder Reactor (LMFBR) fuel cycle are likely to produce population dose commitments small compared to those produced by naturally occurring alpha emitters and globally dispersed transuranium radioisotopes from tests of nuclear weapons in the atmosphere. Potential health consequences of these releases to current and future generations are estimated to be very small compared to risks associated with the production of energy by fossil fuels. The estimates are subject to a number of uncertainties imposed by lack of knowledge. Some of the uncertainties are not likely to be greatly reduced until LMFBR facilities are designed and operated. Others may be significantly reduced prior to facility design and operation. The paper discusses the sensitivity of the estimates to uncertainties and approaches to reducing those uncertainties that strongly influence the estimates.

- 5266 **NUCLEAR POWER GENERATION PLAN IN FRANCE.** Dejou, A. (Electricite de France, 75 - Paris). Genshiryoku Chosa Jiho; No. 9, 77-90(1976). (In Japanese).

From 9. annual conference of the Atomic Industrial Forum; Tokyo, Japan (10 Mar 1976).

The nuclear power generation plan in France was drawn up by taking the prediction of economical growth and power consumption, the capital cost and fuel cost, and also the reduction of the dependence on foreign energy resources and the saving of foreign currency into consideration. The nuclear power generation plan in France started with natural uranium-gas-graphite type reactors, and at present, 8 reactors are in operation. The policy of interrupting the development of this type of reactors and changing to PWRs was decided because of economical reason and the intention to advance into foreign markets. BWRs were abandoned because the cost of development was too expensive and the division of the domestic market should be avoided. The nuclear power stations of 900,000 kW being planned now comprise 32 reactors, and 6 of them were already ordered. 16 reactors will start the operation from 1979 to 1981. The light water reactors of 1.3 million kW will be adopted from 1982. Also the development of liquid metal FBRs was decided. In 1985, the power consumption will be 385 billion kWh, and the nuclear power will amount to about 49%. The policy of siting nuclear power stations in France is explained.

- 5267 **ERBR-II: MORE THAN A DECADE OF EXPERIENCE WITH AN EXPERIMENTAL LMFBR POWER**

PLANT. Staker, R.G.; Cissel, D.W.; Smith, R.N.; Smith, R.R. (Argonne National Lab., Idaho Falls, ID). Proc. Am. Power Conf.; 38: 237-245(1976).

From American power conference; Chicago, IL, USA (20 Apr 1976).

See CONF-760469--.

Experience from the 12 successful years of EBR-2 operation is reviewed. Topics covered include plant description, plant operations, plant maintenance, performance and reliability, and the irradiation program.

- 5268 **PHENIX POWER STATION EXPERIENCE.** Megy, J.M.; L-Zaleski, C.P. Proc. Am. Power Conf.; 38: 246-255(1976).

From American power conference; Chicago, IL, USA (20 Apr 1976).

See CONF-760469--.

Plant characteristics, construction and start-up experience, and operating experience for the Phenix Reactor are reviewed.

- 5269 **OBJECTIVES AND OVERVIEW: THE CLINCH RIVER BREEDER REACTOR PLANT.** Caffey, L.W.; Van Nort, P.S.; Copeland, R.L. (Project Management Corp., Oak Ridge, TN). Proc. Am. Power Conf.; 38: 256-264(1976).

From American power conference; Chicago, IL, USA (20 Apr 1976).

See CONF-760469--.

The objectives, schedule, reference design, management plan, resource requirements, and licensing issues for the CRBR project are reviewed.

- 5270 **RELATIVE CAPITAL COST OF THE LMFBR.** Levenson, M. (Electric Power Research Inst., Palo Alto, CA); Murphy, P.M.; L-Zaleski, C.P. Proc. Am. Power Conf.; 38: 265-269(1976).

From American power conference; Chicago, IL, USA (20 Apr 1976).

See CONF-760469--.

The construction cost estimate for the Clinch River Breeder Reactor is examined in the context of actual cost experience of the nuclear industry for LWR and sodium-cooled plants.

- 5271 **REACTOR SAFETY.** Draper, E.L. Jr. (Univ. of Texas, Austin). pp 43p, Chapter II of Proceedings of National Energy Forum. Volume I. Jackson, J.L. (ed.). Akron, OH; University of Akron (1976).

From National energy forum; Akron, OH, USA (26 Jul 1976).

See CONF-760761--P1.

Information is presented concerning basic reactor kinetics and design characteristics of PWR, BWR, HTGR, and LMFBR type reactors. Reactor protection systems, engineered safeguards, and design basis accidents are described.

- 5272 **DOPPLER REACTIVITY UNCERTAINTIES AND THEIR EFFECT UPON A HYPOTHETICAL LOF ACCIDENT.** Malloy, D.J. Lafayette, IN; Purdue Univ. (1976). 228p. University Microfilms Order No. 77-15,440.

Thesis (Ph. D.).

The statistical uncertainties and the major methodological errors which contribute to the Doppler feedback uncertainty were reviewed and investigated. Improved estimates for the magnitudes of each type of uncertainty were established. The generally applied reactivity feedback methodology has been extended by explicitly treating the coupling effect which exists between the various feedback components.

The improved methodology was specifically applied to the coupling of Doppler and sodium void reactivities. In addition, the description of the temperature dependence of the Doppler feedback has been improved by the use of a two-constant formula on a global and regional basis. Feedback and coupling coefficients are presented as a first comparison of the improved and the currently applied methods. Further, the energy release which results from hypothetical disassembly accidents was simulated with a safety response surface in the parametric safety evaluation code PARSEC. The impact of the improved feedback methodology and of Doppler coefficient uncertainties was illustrated by the usual parametric relationship between available work-energy and the Doppler coefficient. The work-energy was calculated with the VENUS-II disassembly code and was represented as a response surface in PARSEC. Additionally, the probability distribution for available work-energy, which results from the statistical uncertainty of the Doppler coefficient, was calculated for the current and the improved feedback methodology. The improved feedback description yielded about a 16 percent higher average value for the work-energy. A substantially larger increase is found on the high-yield end of the spectrum: the probability for work-energy above 500 MJ was increased by about a factor of ten.

- 5273 SPATIALLY CONTINUOUS APPROACH TO THE DESCRIPTION OF INCOHERENCIES IN FAST REACTOR ACCIDENT ANALYSIS. Luck, L.B. Lafayette, IN; Purdue Univ. (1976). 291p. University Microfilms Order No. 77-15,437.

Thesis (Ph. D.).

The current approach to LMFBR risk assessment is based on a small number of accident path calculations using computer models in which spatial variations are represented with a coarse geometrical mesh, neglecting part of the inter-subassembly incoherence and all intra-subassembly effects. Coherencies in the accident progression resulting from the coarse mesh structure may yield predicted ramp rates significantly higher than more realistic calculations using a finer mesh. Inter-subassembly incoherence can be treated by the use of more channels, but intra-subassembly effects may require a different approach. Future risk assessment is expected to consider, in addition, the effects of modeling and parameter uncertainties. A generalized cell-type approach is developed in which individual subassemblies are represented as a unit. By appropriate characterization of the results of separate detailed investigations, spatial variations within a cell are represented as a superposition. The advantage of this approach is that costly detailed cell-type information is generated only once or a very few times. Spatial information obtained by the cell treatment is properly condensed in order to drastically reduce the transient computation time.

- 5274 MECHANICAL PROPERTIES OF FAST REACTOR FUEL CLADDING FOR TRANSIENT ANALYSIS. Hunter, C.W.; Johnson, G.D. (Hanford Engineering Development Lab., Richland, WA). pp 101-118 of Irradiation effects on the microstructure and properties of metals. Wheeler, J.B.; Hoersch, H.M.; McGlinchey, E.J. (eds.). Philadelphia; American Society for Testing and Materials (1976).

From 8. symposium of the American Society for Testing and Materials on the effect of radiation on structural materials; St. Louis, MO, USA (4 May 1976).

See CONF-760515--.

In order to determine mechanical behavior under various simulated reactor transient events, internally pressurized specimens of fast flux-irradiated 20 percent cold-worked Type 316 stainless steel fuel pin cladding were rapidly heated until they burst. Tests were conducted at heating rates of 100°F/s and 200°F/s with pressures of 2500 to 14,300 psi (17.2 to 98.6 MPa), resulting in failure temperatures from 1000 to 2000°F (538 to 1093 K). The specimens were taken from subassemblies irradiated in the Experimental Breeder Reactor-II at temperatures from 700 to 1300°F (371 to 705 K). Peak burnup and fluence levels ranged from 28,000 to 50,000 megawatt days per metric ton metal (MWD/MTM) and 2.5 to 4.0 x 10²² neutrons (n)/cm² (E > 0.1 MeV), respectively. Irradiation decreased both the failure strain and failure strength, when the transient test conditions resulted in intergranular fracture; intergranular fracture occurred above 1000 to 1200°F (538 to 649 K), depending on strain rate. Below these temperatures the fracture mode is transgranular, so that the failure strength is not reduced. A correlation based on the ratio of irradiated material failure strain and failure strength was developed to describe the effects of irradiation on the mechanical properties of the cladding. The strain ratio correlation indicates that the failure strain does not further decrease beyond a fluence of 2 x 10²² n/cm². The mechanical properties of cladding which had contained fuel during irradiation were degraded more than were the properties of unfueled material.

- 5275 TENSILE PROPERTIES OF FAST REACTOR IRRADIATED TYPE 304 STAINLESS STEEL. Fish, R.L.; Hunter, C.W. (Hanford Engineering Development Lab., Richland, WA). pp 119-138 of Irradiation effects on the microstructure and properties of metals. Wheeler, J.B.; Hoersch, H.M.; McGlinchey, E.J. (eds.). Philadelphia; American Society for Testing and Materials (1976).

From 8. symposium of the American Society for Testing and Materials on the effect of radiation on structural materials; St. Louis, MO, USA (4 May 1976).

See CONF-760515--.

Tension tests were performed on annealed Type 304 stainless steel specimens sectioned from Experimental Breeder Reactor-II (EBR-II) duct thimbles. The specimens had accumulated neutron fluences of up to 10.3 x 10²² neutrons (n)/cm², E > 0.1 MeV (10.3 x 10²⁶ n/m², E > 16 fJ). Irradiation temperatures for the material tested ranged from 700 to 750°F (371 to 405 K). The tests were performed at temperatures of room temperature to 1400°F (773 to 705 K), employing strain rates of 2 x 10⁻³ to 2/min (3.3 x 10⁻⁵ to 3.3 x 10⁻²/s). At test temperatures below 800°F (700 K), the strength increased with fluence until about 7 x 10²² n/cm² (7 x 10²⁶ n/m²), beyond which no further increase was observed. The elongation decreased with fluence, reaching levels as low as 0.5 percent uniform and 1.3 percent total elongation at 700°F (644 K). High fluence failures below 800°F (700 K) occurred by transgranular channel fracture. At test temperatures above 1000°F (538 K), elongations were reduced to very low levels at high fluences (average total elongation (TE) = 0.03 percent). Extensive intergranular fracture at the high fluences produced the low ductility and resulted in failure of the specimens before reaching the strength characteristic of the true hardness of the material.

- 5276 DEVELOPMENT AND EVALUATION OF A STRESS-FREE SWELLING CORRELATION FOR 20 PERCENT COLD-

WORKED TYPE 316 STAINLESS STEEL. Garner, F.A.; Laidler, J.J.; Guthrie, G.L. (Hanford Engineering Development Lab., Richland, WA). pp 208-226 of Irradiation effects on the microstructure and properties of metals. Wheeler, J.B.; Hoersch, H.M.; McGlinchey, E.J. (eds.). Philadelphia; American Society for Testing and Materials (1976).

From 8. symposium of the American Society for Testing and Materials on the effect of radiation on structural materials; St. Louis, MO, USA (4 May 1976).

See CONF-760515--.

Sufficient data were accumulated on the swelling behavior of 20 percent cold-worked Type 316 stainless steel to make predictions of the anticipated swelling in the fast test reactor (FTR). A predictive stress-free swelling correlation was developed employing a limited data base developed in the Experimental Breeder Reactor-II (EBR-II). Insight gained from electron and ion simulation experiments aided in the correlation development. Application of this correlation to environments other than fast reactor core regions requires consideration of the differences in neutron spectrum and displacement rate in such environments.

5277 MICROSTRUCTURAL CHANGES IN ION-IRRADIATED COMMERCIAL ALLOYS. Bell, W.L. (General Electric Co., Pleasanton, CA); Lauritzen, T.; Darlin, E.S.; Warner, R.W. pp 353-369 of Irradiation effects on the microstructure and properties of metals. Wheeler, J.B.; Hoersch, H.M.; McGlinchey, E.J. (eds.). Philadelphia; American Society for Testing and Materials (1976).

From 8. symposium of the American Society for Testing and Materials on the effect of radiation on structural materials; St. Louis, MO, USA (4 May 1976).

See CONF-760515--.

The effect of heavy ion bombardment on the microstructural stability of several high nickel superalloys was evaluated by transmission electron microscopy techniques. The work was intended to simulate the type and magnitude of radiation damage occurring with high fluence exposure in a liquid metal fast breeder reactor (LMFBR) core. Results revealed a general acceleration in precipitation kinetics in alloys with high titanium-to-aluminum ratios, over the temperature range 475 to 725°C (887 to 1337°F). In the Inconel alloys 706 and 718, for example, bombardment produced the stable overaging precipitate η Ni_3Ti at the expense of the preexisting phases γ' , γ'' , and δ . Thermal control data over the same temperature range did not reveal the presence of η . Little or no changes in precipitate microstructure were observed in Nimonic PE16, an age-hardenable alloy with a titanium/aluminum ratio of unity.

5278 PHENIX CONSTRUCTION AND OPERATING EXPERIENCE. Megy, J.M.; Conte, F.; Goddet, J.; Noel, H. pp 3-12 of 1976 ASME-ANS international conference on advanced nuclear energy systems. New York; American Society of Mechanical Engineers (1976).

From ASME-ANS joint international conference on advanced nuclear energy systems; Pittsburgh, PA, USA (13 Mar 1976).

See CONF-760303--.

The methods and principles adopted for the construction of Phenix, which was put into service virtually on schedule and at a cost approaching the initial estimate, are discussed. By the end of 1975, Phenix had already proved a valuable experience, since this power plant has been operating

industrially since July 14, 1974, with availability greater than 73%. The operating results significantly confirm the options adopted for the project, and particularly the pool concept. Moreover, Phenix is a clean power plant, with almost undetectable gaseous wastes, a very low quantity of liquid wastes, and very low thermal wastes, thanks to its high efficiency (over 45%). The experience thus gained provides a solid base for the forthcoming construction of Super Phenix.

5279 SNR CONSTRUCTION EXPERIENCE. Brandstetter, A.; Guthmann, E.A.; Stoehr, K. (Internationale Natrium-Brutreaktor-Bau Gesellschaft mbH, Bensberg, Ger.). pp 97-103 of 1976 ASME-ANS international conference on advanced nuclear energy systems. New York; American Society of Mechanical Engineers (1976).

From ASME-ANS joint international conference on advanced nuclear energy systems; Pittsburgh, PA, USA (13 Mar 1976).

See CONF-760303--.

Construction of SNR 300 began 1973 at Kalkar, lower Rhine. The licensing procedure led to a delay of more than one year. The concrete work is nearly completed, component fabrication is well underway, the reactor vessel already has been pressure tested. Beginning of preoperational tests is expected for 1979.

5280 SNR COOLANT SYSTEM COMPONENTS. De Haas Van Dorsser, A.H. (Neratoom, The Hague, Netherlands); Mausbeck, H. pp 181-191 of 1976 ASME-ANS international conference on advanced nuclear energy systems. New York; American Society of Mechanical Engineers (1976).

From ASME-ANS joint international conference on advanced nuclear energy systems; Pittsburgh, PA, USA (13 Mar 1976).

See CONF-760303--.

The DEBENELUX prototype fast reactor power plant SNR 300 at Kalkar has a loop-type heat transfer system similar to that of the prototype LMFBR plants in the USA and Japan. There exist three 257 MW/sub th/ primary sodium loops, each with a hot leg centrifugal pump and three 85.6 MW/sub th/ intermediate heat exchangers in parallel. From there the heat is transferred to the steam generators via three secondary sodium loops with one cold leg sodium circulating pump in each. At a nominal reactor outlet temperature of 819°C and a turbine inlet power of 771 MW/sub th/ super heated steam of 166 bar and 733°C is produced, giving rise to a plant rating of 327 MW/sub e/ gross. The primary and secondary loops are described in detail.

5281 LMFBR COOLANT SYSTEM COMPONENTS DEVELOPMENT IN JAPAN. Nakai, Y.; Mochizuki, K.; Aoki, T.; Yamaki, H. pp 193-213 of 1976 ASME-ANS international conference on advanced nuclear energy systems. New York; American Society of Mechanical Engineers (1976).

From ASME-ANS joint international conference on advanced nuclear energy systems; Pittsburgh, PA, USA (13 Mar 1976).

See CONF-760303--.

A review is made on the major components and the research and developments (R and D) of them in the coolant system of sodium-cooled fast reactors in Japan. Two fast reactors, the experimental fast reactor 'Joyo' and the prototype fast breeder reactor 'Monju', are being developed as the national project by the Power Reactor and Nuclear Fuel Development Corporation (PNC). In accordance with the project, design, research and development have

been extensively made by PNC itself and industries, including in-sodium and water tests of prototype components, characteristic tests, fabrication tests, etc. for both reactors. "Joyo" with initial output of 50 Mwt is now in preoperational test, while "Monju" with 300 MWe is in the stage of check and review by the governmental authority.

5282 US LMFBR COOLANT SYSTEM COMPONENTS.

Skavdahl, R.E. (General Electric Co., Sunnyvale, CA); Buggy, J.J.; Barratt, R.O. pp 215-235 of 1976 ASME-ANS international conference on advanced nuclear energy systems. New York; American Society of Mechanical Engineers (1976).

From ASME-ANS joint international conference on advanced nuclear energy systems; Pittsburgh, PA, USA (13 Mar 1976).

See CONF-760303--.

Development and use of LMFBR coolant system components in the United States has progressed through more than two decades, with both the government and private sectors participating significantly in the support and conduct of the work. The purpose of this paper is to present the major functional requirements that these components are expected to meet, their current status and future direction of development, and recent test results. The components to be discussed are: (1) primary and intermediate system large sodium piping; (2) primary and intermediate system large sodium valves; (3) primary and intermediate main sodium pumps; (4) intermediate heat exchanger (IHX); and (5) sodium-heated steam generators. The major near-term activities associated with the above components are focused primarily on the Fast Flux Test Facility (FFTF), the Clinch River Breeder Reactor Plant (CRBRP), and the anticipated commercial LMFBR's.

5283 REACTOR SYSTEM COMPONENTS OF JAPAN'S

FAST BREEDER REACTORS. Nakai, Y.; Hirasawa, Y.; Kishida, K.; Nakano, H. pp 427-444 of 1976 ASME-ANS international conference on advanced nuclear energy systems. New York; American Society of Mechanical Engineers (1976).

From ASME-ANS joint international conference on advanced nuclear energy systems; Pittsburgh, PA, USA (13 Mar 1976).

See CONF-760303--.

Japan's experimental fast reactor JOYO is under the precommissioning test and is expected to go critical in late 1976. The design of the prototype MONJU, based on the original design philosophy and guidelines, is now approaching the final confirmation of the overall design in preparation for the construction permit application. A research and development program (R and D) in accordance with the MONJU design is presently being carried out and some tests have been completed. The results of the R and D program associated with JOYO and the experience gained in the construction of JOYO are extensively reflected on the MONJU design. Both JOYO and MONJU are of the loop type and basically of the same concept. This paper presents the principal design bases and the requirements of the two fast reactors, and the nature of research and development programs of the reactor key components.

5284 CLINCH RIVER BREEDER REACTOR PROJECT

FUEL HANDLING DEVELOPMENT PROGRAM. Foster, K.W. (Atomics International, Canoga Park, CA). pp 473-479 of 1976 ASME-ANS international conference on advanced nuclear energy systems. New York; American Society of Mechanical Engineers (1976).

From ASME-ANS joint international conference on advanced nuclear energy systems; Pittsburgh, PA, USA (13 Mar 1976).

See CONF-760303--.

The fuel handling system for the Clinch River Breeder Reactor Project (CRBRP) is based primarily on existing technology and in many respects follows the concept developed for the Fast Flux Test Facility (FFTF). However, several requirements are considerably more stringent and demand more advanced equipment. This paper describes the development program initiated to verify improved design features and performance characteristics.

5285 SUPER-PHENIX: STATUS OF THE CREYS-MALVILLE PLANT.

Casini, Robert; Janberg; Rodet; Megy; Crette; Leduc; Granito. pp 547-562 of 1976 ASME-ANS international conference on advanced nuclear energy systems. New York; American Society of Mechanical Engineers (1976).

From ASME-ANS joint international conference on advanced nuclear energy systems; Pittsburgh, PA, USA (13 Mar 1976).

See CONF-760303--.

The Creys-Malville plant, using a Super-Phenix NSS System, is designed as a basic power plant in the same manner as a light water plant. Main operating conditions, and main technical choices made in order to satisfy these operating conditions, are recalled. Then, the plant is briefly described.

5286 STATUS OF PRELIMINARY DESIGN OF SNR 2.

Guthmann, E.A. pp 563-569 of 1976 ASME-ANS international conference on advanced nuclear energy systems. New York; American Society of Mechanical Engineers (1976).

From ASME-ANS joint international conference on advanced nuclear energy systems; Pittsburgh, PA, USA (13 Mar 1976).

See CONF-760303--.

The SNR 2 is being designed by the Debenelux industry for an international utility association of RWE, EdF and ENEL, the same association that will order SUPERPHENIX. Boundary conditions, R + D, and design features of this large loop type fast breeder power plant, which will follow the SNR 300 after 1980, are discussed.

5287 STUDY OF THE COMMERCIAL FAST BREEDER REACTOR IN JAPAN.

Tanaka, N.; Kojima, T. (Tokyo Electric Power Co. Inc., Japan). pp 571-581 of 1976 ASME-ANS international conference on advanced nuclear energy systems. New York; American Society of Mechanical Engineers (1976).

From ASME-ANS joint international conference on advanced nuclear energy systems; Pittsburgh, PA, USA (13 Mar 1976).

See CONF-760303--.

The research and development (R and D) on the fast breeder reactor (FBR) in Japan are carried on mainly by Power Reactor and Nuclear Fuel Development Corporation (PNC). As the first step, construction and installation of the experimental FBR, "Joyo" at PNC O-arai Engineering Center, have been completed. It is under preoperational function tests, and is scheduled to go critical in mid-1976. As the second step, PNC has performed the R and D for the 300 MWe prototype FBR, "Monju" in preparation for construction. "Monju" is expected to go critical in about 1982. Several large facilities to obtain the technological experience for "Monju" have been installed at O-arai Engineering Center, and some tests have been completed. The R and D for "Monju" have also been progressed by several nuclear

manufacturers and in cooperation with the foreign countries. The FBR is recognized as the reactor type occupying the most important position after the LWR in view of effective utilization of nuclear fuel resources and reduction of separative work. In such consideration, TEPCO contributed in the operation of the Fermi-1 project in the United States, and has been participating in the PNC's activities. In addition to these efforts, to forecast the future situations of nuclear power as a utility, joint studies have been made with four nuclear industry groups since 1969, in order to make clear the concept of the large commercial FBR and the course of its development.

- 5288 OPERATING EXPERIENCE WITH EXPERIMENTAL BREEDER REACTOR-II (EBR-II). Perry, W.H.; Lentz, G.L.; Holmes, J.T.; Wolz, G.C. (Idaho National Engineering Lab., Idaho Falls). pp 29-54 of 1976 ASME-ANS international conference on advanced nuclear energy systems. New York; American Society of Mechanical Engineers (1976).

From ASME-ANS joint international conference on advanced nuclear energy systems; Pittsburgh, PA, USA (13 Mar 1976).

See CONF-760303--.

As originally designed and operated, EBR-II successfully demonstrated the concept of a sodium-cooled fast breeder power plant with a closed fuel reprocessing cycle (mini-nuclear park). Subsequent operation has been as a test and fuel irradiation facility, a role which will continue into the foreseeable future. Since 1961, the operating experience of EBR-II has been very satisfactory. Most of the components and systems have performed well. In particular, the mechanical performance of heat-removal systems has been excellent. A review of the operating experience reveals that all the original design objectives have been successfully demonstrated. To date, no failures or incidents have occurred where serious in-core or out-of-core consequences have resulted. No water-to-sodium leaks have been detected over the life of the plant. At the present time, the facility is operating very well and continuously except for short shutdowns required by maintenance, refueling, modification and minor repair. A plant factor of 66.1 percent was achieved for the calendar year 1975.

- 5289 FAST FLUX TEST FACILITY MAJOR COMPONENT TESTING EXPERIENCE AND PLANT CONSTRUCTION EXPERIENCE. Mairson, R.C.; Neuman, M.R.; Rutkauskas, V.J.; Shoemaker, F.M.; Tollbom, L.R.; Trego, L.A. (Westinghouse Hanford Co., Richland, WA). pp 55-88 of 1976 ASME-ANS international conference on advanced nuclear energy systems. New York; American Society of Mechanical Engineers (1976).

From ASME-ANS joint international conference on advanced nuclear energy systems; Pittsburgh, PA, USA (13 Mar 1976).

See CONF-760303--.

This paper presents the experiences gained to date in the testing of major reactor and plant prototype components, and in the construction of the Fast Flux Test Facility (FFTF). Several new sodium facilities were constructed and confirmation tests on the prototype components are continuing. Test conditions and results to date are reported for the primary pump, intermediate heat exchanger, sodium-to-air dump heat exchanger, large and small sodium valves, purification cold trap, in-vessel handling machine, instrument tree, control rod system, closed loop ex-vessel machine, refueling equipment and selected

maintenance equipment. The construction experiences described were generally selected to cover the significant differences from prior experiences on light water reactor plants.

- 5290 SAFETY ANALYSIS OF THE FAST BREEDER REACTOR CONFINEMENT MAIN PROBLEMS: APPLICATION TO SUPER PHENIX. Meyer-Heine, A. (Commissariat a l'Energie Atomique, Cadarache, France); Penet, F.; Chalot, A. pp 321-325 of 1976 ASME-ANS international conference on advanced nuclear energy systems. New York; American Society of Mechanical Engineers (1976).

From ASME-ANS joint international conference on advanced nuclear energy systems; Pittsburgh, PA, USA (13 Mar 1976).

See CONF-760303--.

This paper will be limited to the safety analysis problems of a fast reactor as far as large accidents are concerned. It will deal, more specifically with accidents which can happen in the core, and will not take into account accidents such as seisms, floods, or large sodium fires. Trying to be as general as possible, it is shown, taking Super Phenix as an example, how some specific problems of the fast breeders can be solved. The main difficulties involved in the confinement specifications will be exposed in the first part of the paper. The analysis of a highly hypothetical accident used to specify the confinement is performed.

- 5291 SAFETY DESIGN CRITERIA FOR LIQUID METAL FAST BREEDER REACTORS. Keaten, R.W. (Atomics International, Canoga Park, CA); Bradbury, P.; Hikido, K. pp 327-336 of 1976 ASME-ANS international conference on advanced nuclear energy systems. New York; American Society of Mechanical Engineers (1976).

From ASME-ANS joint international conference on advanced nuclear energy systems; Pittsburgh, PA, USA (13 Mar 1976).

See CONF-760303--.

The general safety approach for liquid metal fast breeder reactors (LMFBR's) is very similar to that which has been developed for light water reactors. The safety of the plant is assured by design features which prevent the occurrence of accidents, features which limit the damage and serve to protect the public against excessive exposure to radioactive material even in the event of a hypothetical, low-probability accident. Multiple barriers consisting of fuel and cladding, primary coolant boundary, and containment building or other low leakage containment structures are also the same as those utilized for light water reactors. Because of the great similarity in the general approach of LMFBR's and LWR's, the question of safety criteria for LMFBR's is discussed in terms of a comparison with LWR safety criteria.

- 5292 (BNWL-tr--266) THERMAL CONDUCTIVITY AND THERMAL BEHAVIOR OF ABSORBER MATERIALS FOR FAST BREEDERS. Horle, M.; Schulz, B. 1976. Translation of AED-Conf--1976-013-196 (CONF-760359--5). 10p. Dep. NTIS, PC A02/MF A01. From Reactor meeting; Duesseldorf, F.R. Germany (30 Mar 1976).

The thermal conductivity was determined for boron carbide and Eu_2O_3 to 1600 K, and for Eu sub 0.99/ B_6 , to 1200 K. Because of the sensitivity of the thermal conductivity of the latter compound to structure, aspects of metallography and chemical analysis are considered. 5 figures, 2 tables. (RWR)

- 5293 (DOE-tr--28) EDF RESEARCH ON FAST NEUTRON REACTORS. (Electricite de France, 75 - Paris. Direction des Etudes et Recherches). 1976. Translation of French report 6p. Dep. NTIS, PC A02/MF A01.

In order to make possible the calculation of the temperatures of the sodium, of the sheath and of the fuel in fast reactor assemblies, taking into account the mixing phenomena induced by the helicoidal wires, two design codes have been developed. These codes have then been adapted for their integration in the Supercalcyon system. This system shall constitute the reference tool for the development of those codes that shall manage Phenix, and other reactors of the family. Cooling accidents, thermohydraulic studies, and steam generator studies are also in progress.

- 5294 PHOSPHATE-BONDED ZIRCONIA RAMMING COMPOUNDS FOR THE THERMAL STRIPPING OF THE FUEL ELEMENTS OF FAST REACTORS. Pastushkov, V.G.; Novoselov, G.P.; Borisov, G.B. Refractories (USSR) (Engl. Transl.); 17: No. 11/12, 687-691(1976).

An investigation was carried out of the interaction of a phosphate-bonded rammed zirconia compound with molten stainless steel in the presence of (U, Pu)O₂. Trials were carried out in the thermal stripping of oxide fuel elements with a steel shell of a melting and separating unit with a concrete-lined induction coil and a rammed crucible of stabilized zirconia with a binder in the form of orthophosphoric acid. The constructional integrity of the unit remained intact after 75 cycles of thermal stripping. 10 refs.

- 5295 FLUIDIZED BED STEAM GENERATOR CONCEPTS FOR LMFBR. Keairns, D.L. (Westinghouse Research Lab., Pittsburgh); Newby, R.A.; Archer, D.H. pp 569-575 of Fluidization technology. Volume II. Keairns, D.L. (ed.). Washington, DC; Hemisphere Publishing Corp. (1976).

From International fluidization conference; Pacific Grove, CA, USA (15 Jun 1975).

See CONF-7506158--P2.

A fluidized bed steam generator, FBSG, system is being developed for the liquid metal fast breeder reactor, LMFBR. The FBSG eliminates the need for an intermediate sodium loop and heat exchanger utilized in the current design and isolates the sodium and water thus reducing the possibility of a sodium-water reaction. Alternative fluidized bed steam generator concepts are presented and design and operating parameters reviewed. An experimental program is underway to develop design data and demonstrate concept feasibility. The experimental program objectives are reviewed.

- 5296 RESTRAINTS FOR OPERATION OF FBTR STEAM GENERATOR. Vaidyanathan, G.; Kumar, S.S. (Reactor Research Centre, Kalpakkam (India). Fast Breeder Test Reactor Design Group). pp 144-154 of Proceedings of the symposium on power plant dynamics and control. Bombay; Department of Atomic Energy (1976).

From Symposium on power plant dynamics and control; Bombay, India (23 Feb 1976).

See CONF-760265--.

The sodium-heated once-through steam generator is a key component of FBTR plant. A compilation is presented of the restraints placed on the operation of the steam generator in static and dynamic conditions. These restraints arise from the corrosive potential of sodium and water, static and dynamic oscillations, thermal stresses, fluctuations of the nucleate boiling to film boiling transition

point, loss of grid etc. The basic factors characterising these phenomena are discussed.

- 5297 DYNAMIC MODEL OF THE UNITED STATES ENERGY SYSTEM: A TOOL FOR ENERGY R AND D PLANNING. Weyant, J.P. Berkeley, CA; Univ. of California (1976). 223p. University Microfilms Order No. 77-4649. Thesis (Ph. D.).

The benefits attributable to the technology products of alternative energy R and D programs should be evaluated in terms of how well the technologies would contribute as integral elements of the total U.S. energy system (rather than as isolated entities as has typically been done in the past). Thus, this model stimulates the dynamics of the evolution of the total energy system by requiring both existing and new technologies to compete for introduction (i.e., commercialization) on a cost-competitive basis that considers the time phasing of the: (1) retirement of energy conversion facilities, (2) growth in end-use demands, and (3) escalation of the costs of extracting depleting domestic energy resources. This approach contrasts with a static model used by ERDA, wherein assumptions must be made for each future year of interest for: (1) maximum capacity constraints for alternative types of conversion facilities, and (2) the cost of energy resources. This model is used to compare the relative consequences and merits of the technology products from three of ERDA's major energy R and D programs: (1) liquid-metal fast breeder reactors, (2) synthetic fuels derived from coal and oil shale, and (3) improved efficiencies for end-use utilizing devices (e.g., space heaters). It is found that the development of synthetic fuels derived from coal and oil shale is the only alternative (of the three considered) that provides energy independence for the U.S. in the next 50 years. However, the possible collapse of the world oil cartel is shown to pose a major retardant to synfuels commercialization by the private sector until at least the end of the century.

- 5298 HEAT TRANSFER IN HELICAL TUBE SODIUM HEATED STEAM GENERATORS. Duchatelle, L.; de Nucheze, L.; Robin, M.G. (CEN/Saclay, France). pp 269-285 of Future energy production systems. Heat and mass transfer processes. Volume I. Denton, J.C.; Afgan, N.H. (eds.). Washington, DC; Hemisphere Publishing Corp. (1976).

For the 1200 MWe Super Phenix project, the CEA jointly with EDF, has carried out a research and development program for an helically coiled tube steam generator to be manufactured by the Fives-Cail Babcock Company. In the 5 MW CEA test facility at Grand Quevilly an experimental program was performed in order to gain the basic knowledge on heat transfer and pressure drop for boiling water flowing in four helically coiled tubes heated by a counter flow liquid metal, which was badly needed for the design. For the experimental coil to tube diameter ratios of 630 : 20, 810 : 20, 1800 : 20, 2700 : 20 correlations for the predictions of the D.N.B., and the friction coefficient in the high Reynolds number range were obtained. A comparison between experimental results and predictions at various heat ratings shows a fair agreement. In addition it is demonstrated that experimental results compare very well with computer code predictions for a 45 MWt mock-up simulating a liquid metal heated once-through steam generator, with 24 tubes arranged in three layers, which has been tested at the EDF Renardieres facility. All correlations used in our computer code are reported.

- 5299 DIRECT AND INDIRECT HEATING EFFECT IN THE DNB TESTS FOR SODIUM HEATED STEAM GENERATORS FOR LMFBR'S. Cumo, M.; Farello, G.E.; Maciuga, C. pp 287-296 of Future energy production systems. Heat and mass transfer processes. Volume I. Denton, J.C.; Afgan, N.H. (eds.). Washington, DC; Hemisphere Publishing Corp. (1976).

LMFBR's steam generators design require the knowledge of the dryout or burn-out power. These systems help greater heated lengths and lower heat fluxes than reactor core channels, so that most of the existing burn-out correlations fail in predicting the burn-out power. The question arises if the electrically heated test sections are equivalent, for what pertains the dryout power determinations to the actual indirectly heated systems. The testing in actual conditions (sodium water systems) do not permit accurate measurement of the dryout power as is possible in electrically Joule heated systems. The present work shows the difference existing between the two basic boundary heating conditions and provides relations to correct the results obtained in Joule heated systems.

- 5300 INFLUENCE OF FISSION FRAGMENTS AND LIQUID-LIQUID CONTACT ON THE MINIMUM SUPERHEAT FOR BUBBLE NUCLEATION IN SODIUM. Bankoff, S.G. (Northwestern Univ., Evanston, IL). pp 297-303 of Future energy production systems. Heat and mass transfer processes. Volume I. Denton, J.C.; Afgan, N.H. (eds.). Washington, DC; Hemisphere Publishing Corp. (1976).

Claxton and Deltrich have computed superheat thresholds at various saturation temperatures as a function of the energy and type of incident radiation. The synergistic effects are considered that may result when a liquid-liquid interface is subjected to fission fragment bombardment, which is the most effective type of radiation for inducing bubble nucleation.

- 5301 INVESTIGATION OF TWO-PHASE LIQUID-METAL MAGNETOHYDRODYNAMIC POWER SYSTEMS. Amend, W.E. (Argonne National Lab., IL); Fabris, G.; Cutting, J. pp 305-317 of Future energy production systems. Heat and mass transfer processes. Volume I. Denton, J.C.; Afgan, N.H. (eds.). Washington, DC; Hemisphere Publishing Corp. (1976).

Magnetohydrodynamic power systems are receiving renewed interest as a result of the need to develop more efficient energy conversion systems. One of these systems, the two-phase Liquid-Metal MHD (LMMHD) system, is under development at the Argonne National Laboratory. The result of the detailed cycle analysis and systems studies, a description of the experimental facility and the basic results obtained, and the thermal and magneto fluid mechanics problems encountered in two-phase liquid-metal flows in the presence of a magnetic field are presented. Detailed cycle analysis and in-depth modeling studies have indicated that power cycles utilizing two-phase liquid metal MHD generators are attractive alternatives to conventional power-generation techniques from the standpoint of coupling to available heat sources and to provide improved efficiency. The studies indicate that the LMMHD cycle will operate efficiently in the temperature range of 1000--1600°F (50% efficiency with a maximum cycle temperature of 1600°F) and is therefore potentially compatible with many advanced heat sources under development such as the LMFBR, fluidized-bed coal combustor, HTGCR and the fusion reactor. Of special interest is the coupling to the

LMFBR thereby eliminating the costly, potentially hazardous liquid-metal/water interface. The results of detailed parametric studies of the heat transfer interfaces between an LMMHD power cycle and an LMFBR and a steam bottoming plant are described. It is shown that the interfaces do place restrictions on the possible range of operating parameters but that a viable system can be designed to couple with an LMFBR operating with a maximum coolant temperature at 1200°F to yield an overall conversion efficiency approaching 45%. Results of the cycle studies and thermodynamic analysis show that the two-phase generator approaches the operation of an infinite reheat turbine and is responsible for the high cycle efficiencies.

- 5302 PROBLEMS OF DISMANTLING, TREATING AND ANALYZING HIGHLY IRRADIATED FUEL ELEMENTS. Lefort, G.; Hayet, L.; Regnaud, F. (CEA Centre d'Etudes Nucleaires de Fontenay-aux-Roses, 92 (France)); Allain, A. (Compagnie d'Ingenierie pour les Reacteurs au Sodium (CIRNA), Centre d'Etudes Nucleaires de Saclay, 91 = Gif-sur-Yvette (France)); Calame-Longjean, A. (CEA Centre de Marcoule, 30 = Bagnols-sur-Ceze (France)); Moreau, C. (Ateliers et Chantiers de Bretagne (ACB), 44 = Nantes (France)). pp 119-140 of Design of and equipment for hot laboratories. Vienna; IAEA (1976). (In French)

From Symposium on the design of and equipment for hot laboratories; Otaniemi, Finland (2 Aug 1976).

For some years hot laboratories have been receiving highly irradiated fuel elements from experimental reactors and new-generation power reactors. The hazards associated with these fuel elements are greater than those encountered in the past and it has been necessary to re-equip numerous facilities, improving the techniques used and the precautions taken in respect of personnel so that the operating conditions remain satisfactory. The operation of well-designed facilities doing this type of work has in fact become safer. Four representative facilities are described as examples - a dismantling cell with an inert atmosphere, associated with the Phenix reactor, a mechanical chopper cell for oxide fuel elements installed at the head of a pilot reprocessing plant, a radiometallurgy laboratory handling breeder fuel elements and californium, and a facility for chemical analysis of fuel elements. The paper shows that proper control has been established over the different hazards inherent in this type of facility in normal operation and in emergency conditions. These results are confirmed by the conditions of operation with teams having very long experience.

- 5303 (BNWL-tr--317) ONE YEAR OF PHENIX OPERATION. Conte, F. Translated from Ann. Nucl. Energy; 3: 85-94(1976). 34p. Dep. NTIS (US Sales Only), PC A03/MF A01.

The report describes the chronological order of operations from the time that the Phenix Power Plant was first connected to the public supply grid on December 13, 1973. The end of tests prior to placement of the plant in service is described briefly. These tests were interrupted by periods for jobs which ended on July 14, 1974. From that date the operating level was appreciable, and the sequence of operations proceeded without major incident up to the present. This chronological description is accompanied by progress diagrams up to the end of March 1975, by production statistics, by the main operating characteristics under different conditions and by a summary of the incidents which occurred during operation.

These factors lead to load and availability ratios of 76% and 81%, respectively, with an overall efficiency above 45%.

- 5304 (RISLEY-Trans--3066) PROGRESS REPORT ON THE CONSTRUCTION OF KALKAR NUCLEAR POWER STATION AND THE PLANNING OF SNR-2. Traube, K. Translated from Atoomenerg. Haar Toepass.; 18: No. 7/8, 168-178(1976). 18p. British Library Lending Div., Yorkshire, England.

Portions of document are illegible.

Progress at the Kalkar Nuclear Power Station is described. The problems of the licensing procedure for this prototype reactor are discussed. As in previous years, the licensing and Safety Assessment procedure has led to further delays. As in previous years, the cause of this is not so much in inherent difficulties of the safety demonstrations for an FBR power station, but rather that the certificates and Safety Assessments necessary for building progress depend on detailed demonstrations, and such Safety Assessments can hardly be based on forerunners. Since commencement of building, this has led to an almost impossible number of conflicting situations resulting in changes in planning, delays and additional expenditure, which probably find a parallel only in the construction of the THTR prototype in Schmehausen, also burdened with similar difficulties.

- 5305 TOPOGRAPHICAL, MICROSTRUCTURAL, AND CHEMICAL EFFECTS OF SIMULATED HIGH HEAT FLUX ON THE CORROSION OF TYPE 316 STAINLESS STEEL IN FLOWING SODIUM. Babnall, C.; Shiels, S.A.; Orbon, S.J. Microstruct. Sci.; 4: 23-25(1976).

Earlier work, performed in Na test facilities around the world, have produced corrosion data for Type 316 SS which has been used to estimate metal loss rates from the core of a liquid metal fast breeder reactor. Practical difficulties precluded evaluation of the influence on corrosion of irradiation, a high heat flux and the high axial temp. gradient the latter produces. Results of corrosion experiments conducted in a test facility designed to produce a temp. rise in the fluid equivalent to a heat flux in the primary heat of approx. 0.5×10^6 Btu/h-ft²) are presented. The axial temp. gradient was such that the Na temp. was increased over a distance of 3 ft from 820°F (440°C) at the inlet, to 1360°F (740°C) at the outlet. The secondary ion mass spectrometer, the electron probe microanalyser, the scanning electron microscope and the optical microscope were used individually to characterize the various compositional, topographical, and structural changes that occurred in the stainless steel as a result of the Na exposure. Several important effects attributable to the high axial temp. gradient were noted.

- 5306 RATCHETTING AND CREEP-FATIGUE EVALUATION FOR NOZZLE-TO-CYLINDER INTERSECTION. Barsom, R.S.; Loomis, R.W.; Stewart, B.D. (Combustion Engineering, Inc., Windsor, CT). pp 57-75 of 1976 ASME--MPC symposium on creep-fatigue interaction. Curran, R.M. (ed.). New York, NY; American Society of Mechanical Engineers (1976).

From Winter meeting of the ASME; New York, NY, USA (5 Dec 1976).

An analytical investigation of the applicability of simplified ratchetting and creep-fatigue rules to liquid metal fast breeder reactor (LMFBR) component geometries is being conducted under the Oak Ridge National Laboratory's LMFBR High-Temperature Structural

Design Program. Both the detailed inelastic rules and the simplified elastic rules are applied to the results obtained from three-dimensional finite element analyses of a nozzle-to-cylinder intersection. The results of both evaluations are compared at several locations on the surface, and an assessment of the degree of conservatism of the simplified methods is discussed.

- 5307 ELEVATED TEMPERATURE FATIGUE: DESIGN-APPLICATION PERSPECTIVE WITH DIRECTIONS FOR IMPROVEMENT. Severud, L.K. (Westinghouse Hanford Co., Richland, WA). pp 41-55 of 1976 ASME--MPC symposium on creep-fatigue interaction. Curran, R.M. (ed.). New York, NY; American Society of Mechanical Engineers (1976).

From Winter meeting of the ASME; New York, NY, USA (5 Dec 1976).

Concerns of design-analysts about the present elevated temperature fatigue design criteria, methods, experimental basis, and application difficulties are presented. The present methods and rules used for evaluating elevated temperature fatigue life of LMFBR nuclear power plant components to satisfy ASME Code requirements are surveyed. Typical operating environments, loading conditions, analysis ingredients and procedures for design of these components are presented. Examples of inelastic time-dependent fatigue design analysis applications are discussed to show the translation of material fatigue specimen properties to the prediction of cyclic life of a nuclear component. Areas of interest to the design-analyst about the form and utilization of the present elevated temperature fatigue rules and criteria are identified with specific recommendations for further work and directions for improvement.

- 5308 FINITE ELEMENT FORMULATION FOR FLUID-STRUCTURE INTERACTION IN THREE-DIMENSIONAL SPACE. Kulak, R.F. (Argonne National Lab., IL). pp 21-37 of Dynamics of fluid-structure systems in the energy industry. Au-Yang, M.K.; Brown, S.J.; McDermott, J.R. (eds.). New York, NY; American Society of Mechanical Engineers (1976).

A development is presented for a three-dimensional hexahedral hydrodynamic finite-element. Using trilinear shape functions and assuming a constant pressure field in each element, simple relations were obtained for internal nodal forces. Because the formulation was based upon a rate approach it was applicable to problems involving large displacements. This element was incorporated into an existing plate-shell finite element code. Diagonal mass matrices were used and the resulting discrete equations of motion were solved using explicit temporal integrator. Results for several problems were presented which compare numerical predictions to closed form analytical solutions. In addition, the fluid-structure interaction problem of a fluid-filled, cylindrical vessel containing internal cylinders was studied.

- 5309 FUEL ASSEMBLY FOR NUCLEAR REACTOR. Betts, C.; Collinson, A.E.; Hart, D.F. (to UKAEA). French Patent 2,296,918/A/. 30 Dec 1975. Priority date 31 Dec 1974, United Kingdom of Great Britain and Northern Ireland (UK), 10p. (In French).

The present invention is intended for LMFBR type reactors. The fuel assembly according to the invention comprises an impedance suppressing the cavitation in the axial flow; the impedance consists of a tubular body

containing a series of plates for defining the flow, arranged in perpendicular relationship to the longitudinal axis of the body. In addition to the cavitation suppressing impedance, the fuel assembly comprises a fuel bundle located inside a tubular jacket extending the tubular body at one end, hinge pins being provided at the other end of the body for locating the fuel assembly inside the reactor core. The fuel assemblies are arranged side by side onto a grid and the coolant is circulated upwards through the jackets. The variable impedances are sturdy enough to endure the vibration forces generated from coolant circulation through narrow passages.

- 5310 IMPROVEMENTS TO BREEDER REACTOR STRUCTURES CONTAINING A HOT LIQUID METAL. Chevallier, R.; Sauvage, M. (to Alsthom, 75 - Paris (France); CEA, 75 - Paris (France)). French Patent 2,337,408/A/. 29 Dec 1975. 9p. (In French).

Description is given of an improvement to liquid metal cooled breeder nuclear reactors and composed of a main vessel composed of a central internal vessel containing the core and the hot liquid metal. This internal vessel is connected to individual cylindrical containments each containing a heat exchanger. Opposite the hot liquid metal flow through holes, the surfaces of the internal vessel and of the cylindrical containments are given shapes constituted by an axisymmetrical surface generated by an arc. Further, the cylindrical containments are secured directly to the internal vessel at that place by joining these two surfaces by a toroidal surface with an outward facing concavity.

- 5311 NEUTRON SHIELDING BLOCK. Goto, Y. (to Mitsubishi Atomic Power Industries, Inc., Tokyo (Japan)). Japanese Patent 1977-77,999/A/. 24 Dec 1975. 3p. (In Japanese).

A method is described to provide a neutron shielding block which does not require much space and is stable under a high temperature for a long period of time, effective for a wide neutron energy range as a local shielding material and easy to treat of induced radioactivity. The shielding block takes the form of container type. The interior of the container is divided into a plurality of sections by a partition plate, and one of the sections is filled with a thermal neutron absorber, and other sections is filled with a neutron moderator. Each section is provided with a vent hole to communicate the interior and exterior and with an air permeable cover for covering the hole.

- 5312 FUEL ASSEMBLY. Tsutsumi, K.; Kato, Y.; Hayashi, T.; Matsushima, H.; Hayashi, H. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-76,592/A/. 22 Dec 1975. 4p. (In Japanese).

The object of the invention is to uniformize the temperature and speed of a coolant at the outlet of a fuel assembly of a fast neutron reactor in order to stabilize the operation of measuring instruments within the reactor. In a fuel assembly in which shields having coolant passages are provided above and below fuel pins, the coolant passages are inclined with respect to the direction of the core axis to permit mixing of the coolant at the outlet of the fuel assembly.

- 5313 EMERGENCY CORE COOLING DEVICE. Kudo, K.; Suzuoki, A.; Goto, T. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-74,797/A/

- 19 Dec 1975. 5p. (In Japanese).

The object of the invention is to remove decay heat produced after a scram without subjecting the cooling system of a nuclear reactor to thermal shock. In the normal operation of the reactor, sodium passing through the low temperature section of a secondary cooling system is partly extracted and stored in a separately provided tank. The stored sodium is poured again into the low temperature section of the secondary cooling system when the cooling capacity thereof is lost.

- 5314 AUXILIARY COOLING SYSTEM FOR A PRIMARY HEAT EXCHANGER WORKING IN A NUCLEAR REACTOR. (to Societe Stein Industrie, 75 - Paris (France)). French Patent 2,335,916/A/. 18 Dec 1975. 5p. (In French).

The description is given of an auxiliary cooling system for the primary heat exchanger of a fast nuclear reactor. This exchanger comprises at least one tube bundle in which the water to be vaporised and superheated flows; a liquid alkaline metal constituting the primary fluid flows around the tubes and inside the casing. The exchanger casing is wrapped in a cladding with air intake and discharge facilities at its ends and this same casing is fitted with auxiliary surfaces for the exchange of heat with the air that can circulate in the cladding. The auxiliary heat exchange surfaces of the casing are points affixed to it, the air intake and discharge systems making it possible to have forced circulation or natural circulation of the air.

- 5315 CONTROL ROD DRIVES. Tanaka, J. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-72087/A/. 12 Dec 1975. 6p. (In Japanese).

A method to perform maintenance and inspection easily is described. This invention is characterized by division into an upper control rod driving device, which requires more in number of maintenance and inspection, and a lower control rod driving device, which requires less in number of maintenance and inspection. That is, the upper control rod driving device is provided with a driving means and the lower control rod driving device is provided with a control rod holding means and a scram accelerating means for accelerating the falling of the holding means at the time of urgent stoppage of reactor. With this construction, each of driving devices can be separated and individually carried by cask and gases to be fed into the scram accelerating device may be used as gases for operating the control rod holding device so that a single source of gas supply will suffice.

- 5316 CORE CATCHER IN A REACTOR. Kudo, K. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-70294/A/. 10 Dec 1975. 6p. (In Japanese).

A method is described to arrange a material, whose specific gravity is greater than molten nuclear fuel material and melting point is lower than temperature of coolant during operation of the reactor, close to a lower outer peripheral portion of a reactor container to positively prevent development of melt-through trouble. When molten nuclear fuel drops on the bottom within a reactor container resulting from occurrence of melting trouble in a fast breeder, through holes are formed in the bottom of the reactor container. Molten lead in an area of molten lead flows into the pressure vessel of the reactor through the through holes to form a boundary between sodium and molten

lead in a level in which pressure in the reactor container is balanced. At this time, since the molten nuclear fuel is smaller in specific gravity than that of molten lead, the molten nuclear fuel is afloat in the neighbourhood of the boundary to prevent outflow thereof toward outside the reactor container.

- 5317 MATHEMATICAL MODEL OF AN INDUCTIVE TYPE SODIUM LEVEL METER AND SOME APPLICATIONS. Asano, K. (Tokyo Shibaura Electr Co, Kawasaki, Jpn). IEEE Trans. Nucl. Sci.; NS-22: No. 6, 2689-2695 (Dec 1975).

The appearance of a liquid-metal-cooled fast-neutron fuel-breeder reactor has brought various problems to the field of process measurement. The measurement of liquid metal level is one of those problems. A mathematical model of an inductive type liquid sodium level meter has been developed, which simulates behavior of the level meter in various environmental conditions. A test model was made to confirm experimentally the accuracy of the mathematical model. It was recognized that the mathematical model shows fairly well the behavior of an actual level meter. Error estimates were accomplished with this mathematical model. Errors are caused by the change of ambient temperature and by the surface tension of sodium. 8 refs.

- 5318 CONTROL OF WATER LEAKAGE INTO SODIUM BY ACOUSTIC NOISE. Kozlov, F.A.; Sergeev, G.P.; Yugal, V.S. At. Energ. (USSR); 39: No. 6, 432-434 (Dec 1975). (In Russian). ACOUSTIC MONITORING; LEAKS; SODIUM; STEAM GENERATORS; LMFBF TYPE REACTORS; WATER

- 5319 NUCLEAR REACTOR. Aubert, M.; Lambert, P. (to CEA, 75 - Paris (France)). French Patent 2,333,328/A/. 26 Nov 1975. 14p. (In French).

The description is given of an improvement to the cooling system of the main vessel and/or the primary vessel of sodium cooled fast neutron integrated reactors of high power (approximately 1000 MWe). These reactors include a vessel placed inside a containment shell of which the upper part is closed by an upper shield and containing the reactor core, and which is partially filled with liquid sodium, topped with an inert gas blanket. These reactors also have facilities for taking from their upper part the 'hot' liquid sodium after it has been through the core and for re-injecting the cold liquid sodium at the bottom of the vessel and below the core. The vessel includes a first cylindrical ring forming between itself and the vessel a first annular space, one of the two annular spaces serving for the 'cold' liquid metal feed and the second annular space for discharging it. The first ring comprises an upper edge located above the maximum level of the cold liquid inside the discharge annular space. The upper edge is so shaped that the flow of the cold liquid metal from the feed annular space above the edge occurs without the liquid metal flow discharge breaking away from the wall of the first ring.

- 5320 ACCIDENT CONTROL ELEMENT OF SECTIONAL LIQUID METAL STEAM GENERATOR. Srutek, J.; Hlousek, V. Czech Patent 161,417/B/. 15 Nov 1975. 5p. (In Czech).

Sectional steam generators are used in large nuclear power plants with fast liquid metal

cooled reactors. Currently, the individual sections are connected to one compensating vessel in which the control element, e.g., the membrane is located. In case of the failure of one pipe in the section it is consequently necessary to shut down the whole generator. The invention claimed removes this disadvantage by connecting to each section of the generator a compensating vessel on which is mounted a control element, such as for instance a membrane, while a rapid-action valve is placed between the compensating vessels and outlet pipes to the pumps and the intake of sodium from the intermediate heat exchanger and the generator sections. In case of failure of one of the sections it is not necessary to shut down the whole generator but only that particular section while the other sections remain in operation.

- 5321 COVER GAS SYSTEM IN A REACTOR. Iwasaki, T.; Koike, H.; Kawashima, H.; Uemura, A. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-60399/A/. 14 Nov 1975. 5p. (In Japanese).

A tank for temporarily store surplus gases resulting from a pressure differential of cover gases required at the time of power operation and at the time of replacing fuel as the gases processed in a distilling column device for recovering radioactive rare gases is described. A cover gas space in a reactor vessel is connected to a first tank provided with a pressure gage. The cover gas within the first tank is sucked and compressed by a compressor to be stored in a second tank, and then passing through the rare gas removing and recovering device into a cover gas header and thence returned to the aforesaid cover gas space. On the other hand, a third tank is connected to the first tank, said third tank being connected to said cover gas header. Also, a cover gas feeding device is connected to the third tank. In this system, extra gases are stored in the third tank.

- 5322 REACTOR CONTAINMENT VESSEL. Kawashima, H.; Naito, K. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-59289/A/. 10 Nov 1975. 5p. (In Japanese).

A method is described to restrain a rise in pressure within a sealed space in a sodium cooled fast breeder, such rise in pressure being formed within a reactor vessel as a result of failure of coolant. A good thermal conductive member such as a heat pipe is disposed between a plurality of spaced sealing spaces for receiving a cooling system. When sodium leaks within one sealed space to cause a temperature to rise, part of the temperature is directly cooled by means of an atmosphere control system within own chamber while the other part of the temperature is transmitted to the other chamber by means of the heat pipe and is cooled by means of the atmosphere control system therein.

- 5323 DEFECTIVENESS DETECTOR. Michiguchi, Y.; Suzuki, K.; Takahashi, F. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-57498/A/. 7 Nov 1975. 4p. (In Japanese).

A method to permit three-dimensional determination of the position of abnormal sound generation and also permit early and reliable detection of the defectiveness of an apparatus is described. A plurality of wave guides providing different speeds of sound propagation are provided on a plurality of points in a machine at which the defectiveness is to be detected, and each of them is connected to a sound detector. The defectiveness is detected

by determining a point of generation of defectiveness from the time difference in the detection of a defective sound signal in each detector.

- 5324 STRAINER DEVICE. Mokuya, K. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-57499/A/. 7 Nov 1975. 4p. (In Japanese).

A method to provide a strainer device, which is adapted to facilitate flushing and is particularly suited for installation in the cooling system of a liquid metal cooled fast breeding reactor is described. A casing accommodating a strainer and a blind plate for the selection of a flow path is provided at a suitable portion of the duct line. The blind plate is adapted to be rotated by an opening and closing means consisting of a rod, bellows, shaft and so forth. At the time of flushing, the duct line is sealed by the blind plate.

- 5325 VAPOR TEMPERATURE CONTROL DEVICE FOR FAST BREEDER REACTOR PLANT. Morimoto, H.; Fujiwara, T. (to Mitsubishi Heavy Industries Ltd., Tokyo (Japan)). Japanese Patent 1977-57495/A/. 4 Nov 1975. 4p. (In Japanese).

A method to permit the temperature control system of a fast breeder reactor to efficiently control the fluid temperature at the outlet of the heated side is described. A bypass system from the outlet of a secondary system sodium pump to the sodium inlet side of a re-heater is provided, and a flow control valve is provided at an intermediate point in the bypass system. Meanwhile, the control system comprises a system for moving the control rods by means of a subtractor for comparing the output of a vapor temperature detector at the outlet of an over-heater and the output of a temperature setter and a subtractor for producing the set value of the temperature at the outlet of the reactor through a proportional integrator-differentiator from the output of said first-mentioned subtractor and a system for moving a bypass flow control valve by operating said proportional integrator-differentiator controller through a subtractor for comparing the output of the re-heater and the output of the temperature setter.

- 5326 THEORETICAL MODEL WITH VARIABLE MASSES FOR THE MOLTEN FUEL--SODIUM THERMAL INTERACTION IN A NUCLEAR FAST REACTOR. Caldarola, L. (Kernforschungszenz Karlsruhe, Ger). Nucl. Eng. Des.; 34: No. 2, 181-201 (Nov 1975).

The reaction has been assumed to occur in two successive phases, A and B. Phase A. A given number (function of time) of fuel particles come into intimate contact with a given mass of liquid sodium (function of time). The heat transfer process is characterized by good direct thermal contact between the fuel and the liquid coolant, and by a large contact area due to the small size of the particles. Phase B. The heat transfer process is now characterized by a very large contact area, and by thermal contact between the fuel and coolant which becomes increasingly worse with time, due to the formation of a sodium vapour layer at the external surface of the fuel fragments. The model accounts for the time history of the temperatures of each fuel particle by means of the use of specially averaged temperature values. The calculation of the heat transfer coefficient during phase B is based on experimental results. The presence of fission gases can also be taken into account. A size distribution of fuel particles has also been incorporated into the model as well as the effect of friction due to the channel walls and

that of the pressure losses at the outlet of the channel. Numerical evaluations are also included and the results are discussed. Diagrams, equations, graphs, and tables show data.

- 5327 ANALYSIS OF THE DECARBURIZATION AND AGING PROCESSES IN 2 1/4CR-1MO STEEL. Klueh, R.L.; Leitnaker, J.M. (Oak Ridge Natl Lab, Tenn). Metall. Trans., A; 6A: No. 11, 2089-2093 (Nov 1975).

Specimens of 2 1/4-Cr-1 Mo steel that had been decarburized in sodium or aged in an inert atmosphere for 26,500 h at 839 K were metallurgically examined with optical and transmission electron microscopy. The carbide particle density of the decarburized and aged specimens was considerably less after 26,500 h than after 10,000 h, was the density in the aged specimen being considerably greater than that in the decarburized specimen. For the aged specimen, the density decrease was a result of Ostwald ripening, and the total amount of carbide was not noticeably affected. Decarburization, however, resulted in a gradient in carbide density, with the smallest density near the surface. Carbides were electrolytically extracted and identified by X-ray diffraction. From the before-test material, which was in the annealed condition and had a ferrite-pearlite microstructure, 1.4 wt pct carbide was obtained. This sample contained M_3C , $M_{23}C_6$, and M_6C , with $M_{23}C_6$ constituting the major portion. In the aged specimen, 2.3 wt pct of precipitate was extracted and contained 60 pct M_6C , the balance $M_{23}C_6$. The decarburized specimen contained two regions with different carbide contents. The first 20 pct of the 1.6 mm thick (0.063 in.) specimen yielded 1.3 wt pct precipitate, which contained 94 pct M_6C , the balance $M_{23}C_6$. The interior of the specimen contained 1.6 wt pct precipitate, 86 pct M_6C .

- 5328 APPARATUS OF HOLDING MOLTEN FUEL IN A REACTOR. Nagashima, K. (Nippon Atomic Industry Group Co. Ltd., Tokyo). (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-53197/A/. 25 Oct 1975. 3p. (In Japanese).

A method to dispose below a reactor container, a sealed container receiving therein cooling medium and a molten fuel holding mechanism and provided with a cooling mechanism whose top plate is in a shape capable of receiving molten fuel thereby preventing an occurrence of re-criticality of molten fuel caused by accidents and to perform positive cooling is described. The molten fuel fallen through the reactor container due to accidents is diffused and stored on the top plate formed with an annular concave portion of a V-shaped section prevent an occurrence of re-criticality and thereafter falls into the molten fuel holding mechanism while being cooled by sodium resulting from destruction of top plate, causing a neutron absorbing material to absorb the neutron.

- 5329 DUAL CONSTRUCTION DEVICE HAVING A MECHANISM FOR DETECTING LEAKAGE OF SODIUM. Kimura, M. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-52092/A/. 24 Oct 1975. 3p. (In Japanese).

A method to provide a device, of which outer wall is composed of a transparent material, interiorly provided with a sodium emitting heater and exteriorly provided with a phototube, so as to early discover a leakage of liquid sodium is described. When the liquid sodium in the inner wall is leaked into the outer wall through a crack, a fluorescence is

generated by heat from a heater, and the generated fluorescence is detected by means of a phototube to actuate an alarm.

- 5330 DEVICE FOR STARTING A STEAM GENERATOR BY HEATING SODIUM IN A REACTOR. Nakano, H. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-48797/A/. 16 Oct 1975. 5p. (In Japanese).

A method to enhance cooperation between ventilation and steam conditions of turbine and ventilation condition relative to a superheater at the time of starting a plant using a fast breeder, and to enhance safety with respect to failure of heat transmission tubes at the time of start is described. In a device in which steam generated in an evaporator is fed to a high pressure turbine through a superheater and an outlet steam of high pressure turbine is reheated by means of a re-heater and fed into a turbine on the side of low pressure to drive the turbine for power generation, opening and closing valves are mounted on outlet and inlet pipes, respectively, of the heat transmission pipe in the super heater, said outlet and inlet pipes being connected by a bypass pipe. Upstream side of the opening and closing valve on the inlet pipe and the downstream side of the opening and closing valve on the outlet pipe and connected by a bypass pipe in the re-heater and said bypass pipe in the re-heater is provided with a steam heat exchanger to be heated by steam in the outlet of the superheater, and a steam line in an auxiliary boiler is connected to the side of re-heater from the opening and closing valve on the heat transmission pipe in the re-heater.

- 5331 FLUIDIZED BED STEAM GENERATOR. Archer, D.H.; Kearns, D.L. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). French Patent 2,356,109/A/. 15 Oct 1975. Priority date 4 Nov 1974, United States of America (USA), 7p. (In French).

This invention concerns a fluidized bed steam generator used with the liquid metals of breeder reactors. Its purpose is to develop a simple heat exchanger for the safe exchange of heat between the slightly radioactive liquid metal and the water and/or steam without permitting any reaction between the liquid metal and the water and/or steam. This heat exchanger includes a vessel, a first bundle of tubes placed in the vessel and receiving the primary fluid and a second tube bundle placed in the vessel at a certain distance from the first bundle of tubes and designed to receive the secondary fluid. The products, in particle form, are placed in the vessel, in the interval existing between the first and second tubes, and a gaseous means fluidizes the particles to form a fluidized bed in the vessel and ensures good heat transfer between the first and second set of tubes by means of the fluidized bed.

- 5332 VENT MECHANISM. Iwai, S.; Ikarimoto, I. (to Mitsubishi Atomic Power Industries, Inc., Tokyo (Japan)). Japanese Patent 1977-47,196/A/. 13 Oct 1975. 5p. (In Japanese).

The object of the invention is to connect a plurality of cladding tubes with gas generating substances received therein to one gas reservoir thereby preventing an excessive rise in internal pressure of the cladding tubes and preventing gases generated by irradiation from being released into coolant. Gases generated from the gas generating substances within the cladding tubes are gathered in a gas plenum and flown into a gas reservoir through a gas communicating pipe, after which the gases pass

through a gas vent line and are released into a fuel exchanger and gas burner.

- 5333 CONTAINER FOR A LIQUID METAL COOLED REACTOR. Naito, K.; Fukushima, Y. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-43097/A/. 3 Oct 1975. 4p. (In Japanese).

A method is described to prevent combustion of leaked liquid metal by sealing the upper portion of a liquid metal pool which is disposed within a liquid metal duct chamber or the like, the liquid metal drain tube being inserted in the chamber, and the inside of the pool being filled with inert gas while disposing a liquid metal cover member. Structure: Sodium leaking into a primary cooling system duct chamber flows, as it fuses the plug, through a drain duct into a sodium pool section which is sealed with a lid. At this time, sodium causes inert gas filled there to flow out through a gas purge duct, thus causing a sodium cover member to float up and cover the top surface of the incoming sodium.

- 5334 COMPOSITION OF ETA CARBIDE PHASE IN 1 1/4 CR--1MO. Leitnaker, J.M.; Klueh, R.L.; Laing, W.R. (Oak Ridge Natl Lab, Tenn). Metall. Trans., A; 6A: No. 10, 1949-1955 (Oct 1975).

The steel investigated is of the type now projected for use in the steam generator of the demonstration Liquid-Metal Fast Breeder Reactor. The metal-to-carbon ratio in the eta phase in specimens of 2.25Cr-1Mo steel exposed to sodium at 566 C for 26,500 hr was found to approach four, rather than six, a fact which may be influenced by the presence of Si, O, and perhaps Cr. Silicon is present in the eta phase, probably on a metal sublattice, while oxygen is probably present on the metalloid sublattice. The observation is supported by both structural and compositional data from the literature.

- 5335 REACTOR. Miyazawa, T. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-41795/A/. 29 Sep 1975. 6p. (In Japanese).

A method to permit failure detection for the reactor fuel through detection of the nuclear fission product in the gaseous phase or solid phase introduced into liquid metal coolant passing through the reactor body is described. Part of the liquid metal coolant passing through the reactor body is led through the sampling system to the outside of the reactor body and is returned through a tank to the reactor body. A neutron detector having a γ -ray shield function is disposed in the neighborhood of the afore-mentioned tank, whereby delayed neutrons emitted by the nuclear fission product contained in the coolant within the tank are decelerated into thermal neutrons for detection. In addition, gas present in the upper part of the tank is led to a nuclear fission product gas detector provided in the tank for detecting the nuclear fission product gas contained in the gas led to the detector. The operation of the reactor body is controlled in accordance with the output signals from the afore-mentioned detectors.

- 5336 CONTROL ROD ASSEMBLY. Matsushima, H.; Kato, Y.; Tsutsumi, K. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-41793/A/. 27 Sep 1975. 4p. (In Japanese).

A method to quickly shut down the reactor by permitting quick insertion of the control rod into the reactor core in a short period of time

is described. The wall of a guide tube into which a control rod is inserted is provided with through holes. Thus, when the control rod is inserted into the guide tube by momentarily inserting compressed rare gas at the time of emergency shut-down of a fast reactor, sodium below the control rod in the guide tube flows out through the aforementioned through holes. Consequently, the control rod can be quickly inserted.

5337 LIQUID METAL COOLED FAST BREEDER.

Nakai, Y.; Yamagishi, Y.; Koga, T. (to Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Japanese Patent 1977-40296/A/. 26 Sep 1975. 3p. (In Japanese).

A method to increase the operation efficiency of the reactor by making optimum the flow distribution of coolant within a reactor container is described. In a fast breeder having a high pressure plenum communicating with the core fuel, a low pressure plenum communicating with the blanket fuel and a lower plenum communicating with the high pressure plenum, the low pressure plenum and lower plenum are connected together via a duct, and a flow control device is provided on the duct for controlling the flow of liquid metal flowing through the core fuel and blanket fuel. The flow control device comprises an electromagnetic pump, a flow control valve, and electromagnetic flow gauge and a pressure gauge. When the flow rate required for the low pressure plenum is lower than a preset flow rate, the electromagnetic pump is operated to return the undesired quantity of coolant from the low pressure plenum to the lower plenum, thus reducing the flow in the low pressure plenum.

5338 LIQUID METAL COOLED NUCLEAR REACTOR.

Guidez, J.; Jarriand, P. (to CEA, 75 - Paris (France); Alstom, 75 - Paris (France)). French Patent 2,326,011/A/. 26 Sep 1975. 9p. (In French).

The invention concerns a fast neutron nuclear reactor cooled by a liquid metal driven through by a primary pump of the vertical drive shaft type fitted at its lower end with a blade wheel. To each pump is associated an exchanger, annular in shape, fitted with a central bore through which passes the vertical drive shaft of the pump, its wheel being mounted under the exchanger. A collector placed under the wheel comprises an open upward suction bell for the liquid metal. A hydrostatic bearing is located above the wheel to guide the drive shaft and a non detachable diffuser into which at least one delivery pipe gives, envelopes the wheel.

5339 CONTROL ROD FOR FAST BREEDER.

Tsutsumi, K.; Takeda, S.; Yamamoto, M. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-39,096/A/. 22 Sep 1975. 2p. (In Japanese).

The object of the invention is to flatten the power distribution in a fast breeder using hexagonal fuel assemblies by inserting a control rod having Y-shaped sectional profile into the space between adjacent fuel assemblies. A control rod assembly is formed into a Y-shape in cross section so that it can be inserted into a space defined by three adjacent wrapper tubes of the fuel assemblies. For the control rod assembly may be used one, in which a pin containing B₄C accommodated in a stainless steel can is housed in a Y-shaped stainless steel sheath. Since this arrangement of absorbent body is more uniform than the conventional arrangement, the distortion of the power distribution in the direction of the

plane of reactor core caused by the insertion and removal of the control rod can be reduced.

5340 LEAK DETECTOR OF LIQUID SODIUM.

Himeno, Y. (to Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Japanese Patent 1977-36,290/A/. 18 Sep 1975. 3p. (In Japanese).

Object: To arrange a cable core connected to a leakage current detector on the outer wall of piping for liquid sodium, devices or the like and apply a voltage to said core and outer wall to quickly and securely detect the leakage of liquid sodium. Structure: A cable, which is composed of metal coating formed of metal material (copper, steel, stainless, etc.) which is apt to be corroded by reaction products of liquid sodium with water and oxygen in air, and metal oxide (such as magnesium oxide, beryllium oxide, aluminum oxide) as an electric insulator is arranged on the outer wall of pipes or devices. In the event sodium is leaked from the pipes or devices, said metal coating and the insulator are corroded, and the leakage of sodium is sensed by a leakage current detector through the core in the cable.

5341 REACTOR.

Orii, S.; Fujiki, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1977-36,287/A/. 16 Sep 1975. 5p. (In Japanese).

The object of the invention is to securely prevent interference of a fuel handling device with a reactor container even in trouble of an interlock device to enhance maneuverability and safety of the fuel handling device in a sodium cooled fast breeder. A simulator for simulating operation of an arm in the fuel handling device and rotational operation of a rotating shielding plug is provided outside the reactor container in cooperation with the fuel handling device and the rotating shielding plug. The simulator is electrically connected to the fuel handling device and the rotating shielding plug so as to control operation of the fuel handling device and the rotating shielding plug. Whereby, the simulator prevents interference of the arm of the fuel handling device with the reactor container.

5342 SODIUM-WATER REACTION PRODUCT CONTAINER FOR LIQUID METAL COOLED FAST BREEDERS.

Yoshida, K.; Tanaka, M.; Sakata, T. (to Babcock Hitachi K.K., Tokyo (Japan)). Japanese Patent 1977-32,496/A/. 8 Sep 1975. 4p. (In Japanese).

A method is described to provide a sodium-water reaction product container, which can reduce the increase in pressure due to the reaction without causing combustion of a great quantity of hydrogen at one time at the time of a sodium-water reaction accident. Hydrogen produced by the reaction between sodium and water and sodium vapor accompanying hydrogen are separated into sodium and hydrogen by a sodium separator within a reaction product container. Sodium is recovered from a pool exhaust valve at the bottom. The separated high temperature hydrogen gas is cooled by a heat exchanger and is thus reduced in volume, and it is then rendered into hydrides through reaction with metal particles such as Mg, Ti, Zr and so forth.

5343 CONTROL EQUIPMENT FOR EMERGENCY COOLING SYSTEM.

Watanabe, T.; Kuwayama, M. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-29598/A/. 29 Aug 1975. 7p. (In Japanese).

A method to improve the standby state of an emergency reactor core cooling system for a fast breeder reactor and increase the heat removal effect at the time of the start is described. An electromagnetic pump for an emergency reactor core cooling system is excited in reverse phase in the standby state so as to restrict the rate of natural circulation and thus hold the liquid metal at a low temperature. At the time of start, the electromagnetic pump is excited in the right phase from an emergency power source.

- 5344 LIQUID-METAL COOLED FAST BREEDER. Tanaka, N.; Yamagishi, Y.; Koga, T. (to Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Japanese Patent 1977-25,996/A/ 25 Aug 1975. 3p. (In Japanese).

The object of the invention is to combine two kinds of entrance nozzles of neutron shielding assemblies to be capable of varying a distribution of flow rates of a high pressure plenum flow and a lower pressure plenum flow thereby accurately and efficiently cooling core forming elements of a liquid-metal cooled fast breeder. A connecting pipe for receiving therein an entrance nozzle for a neutron shielding assembly extends through a high pressure plenum and a lower pressure plenum down to a second low pressure plenum formed therebelow. The entrance nozzle has two kinds, one passing a coolant within the high pressure plenum only into the neutron shielding assembly, the other passing the coolant also into the second plenum. Thus, these two entrance nozzles may be combined to thereby vary the flow rate of high pressure plenum flow flowing into the core fuel assembly and that of low pressure plenum flow flowing into the blanket fuel assembly.

- 5345 LIQUID METAL COOLED FAST BREEDER REACTOR. Nakai, Y.; Yamagishi, Y.; Koga, T. (to Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Japanese Patent 1977-25997/A/. 25 Aug 1975. 3p. (In Japanese).

A method to permit readily making optimum the distribution ratio between the low and high pressure plenum flows of the coolant by the provision of a low pressure lead-in chamber having a jet pump below the low pressure plenum is described. In a liquid metal cooled fast breeder, a high pressure plenum communicating with the core fuel and a low pressure plenum communicating with blanket core fuel are communicated with a lower plenum having a liquid metal inlet. A low pressure lead-in chamber having a lower opening and provided with a jet pump is formed below the low pressure plenum and is provided with a flow control device. Further, a duct open to the lower plenum is connected to the nozzle of the jet pump.

- 5346 IMPROVEMENT TO THE SECONDARY CIRCUITS OF A NUCLEAR REACTOR. Robin, M. (to CEA, 75 - Paris (France)). French Patent 2,321,750/A/. 22 Aug 1975. 7p. (In French).

The invention concerns an improvement to the design of the secondary systems of a nuclear reactor with a view to making them more compact in order to make a significant saving on the lengths of piping required and on the buildings protecting them, thereby reducing the corresponding capital expenditure. As the steam generator is designed with a central return duct for the cold sodium, the sodium is driven through the system by a pump whose suction is directly coupled to the central duct of the generator. The steam generator and circulating pump unit is mounted in a sealed bay integrated

in the reactor building, the connection with the intermediate exchanger in the secondary system being made by pipes passing through a leak-proof opening in the wall of the building.

- 5347 SYSTEM FOR EXTRACTING AND INSERTING AN IRRADIATION CAPSULE INTO A FUEL ASSEMBLY. Allain, A.; Delamotte, M. (Societe Technique pour l'Energie Atomique (Technicatome), Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)). (to CEA, 75 - Paris (France)). French Patent 2,320,616/A/. 5 Aug 1975. 18p. (In French).

The invention concerns a system enabling all the requisite operations to be carried out to extract or insert an irradiation capsule fitted with a holding head into a fuel assembly able to take such a capsule and in which the capsule can be locked. For such an assembly, the fuel elements are maintained between two coaxial tubes, the inner tube marking a housing for such a capsule and having a groove into which articulated claws on the capsule can engage. This invention applies satisfactorily in the fuel assembly handling system of a fast neutron nuclear reactor.

- 5348 SYSTEM OF EXTRACTION AND INTRODUCTION OF AN IRRADIATION CAPSULE INTO A FUEL ASSEMBLY. Allain, G.; Delamotte, M. (Societe Technique pour l'Energie Atomique (Technicatome), Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)). (to CEA, 75 - Paris (France)). French Patent 2,320,616/A/. 5 Aug 1975. 18p. (In French).

This invention applies in the fuel assembly handling system of a fast neutron reactor. This device enables an irradiation capsule to be extracted from or introduced into a fuel assembly when the latter is maintained vertical in a drum located inside the receptacle for storing new or irradiated assemblies which, located on the periphery of the reactor vessel, is filled with a liquid metal capped with an inert gas. The device can be moved straight up from an assembly. Inside a tubular body that can abut on the assembly in question, it includes a central tube ending with a part capable of coming into contact with the gripping head of the capsule and fitted with a grab; a grab control tube that can unlock the capsule in the assembly and a mechanical system capable of generating successive vertical displacements of the tubular body and the central tube in opposite directions, once the device has been immobilised.

- 5349 SAFETY MARGINS FOR SODIUM-BONDED CARBIDE FUEL. Siegmann, E.R.; Gilbertson, J.C. (Combust Eng Inc, Windsor, Conn). Nucl. Technol.; 26: No. 4, 452-459 (Aug 1975).

The consequences of the loss of bond in a sodium-bonded pin were considered. New calculations indicated that loss of bond could lead to fuel redistribution within the clad but not to clad failure. Fuel-coolant interactions with carbide fuel were considered. Interface temperatures between molten fuel and sodium were determined and applied to the Fauske criterion for spontaneous nucleation. Results indicate that rapid sodium vaporization is not likely with carbide fuel. Upper limit work-energy yields were calculated for sodium contact with molten oxide and carbide fuels using SOCOOL-II code. Temperatures, properties, and geometry were varied to determine the relative dependency of the results.

- 5350 OPTIMUM PIN DIAMETER FOR LMFBR CARBIDE FUELS. Noyes, R.C.; Doherty, P.K.;

Caspersson, S.A.; Hanus, N.; Stuteville, D.W. (Combust Eng Inc, Windsor, Conn). Nucl. Technol.; 26: No. 4, 460-471(Aug 1975).

Subsidiary parameters, including coolant velocity, burnup, and properties of the clad and subassembly duct material, were studied. The materials used were Type 316 stainless steel and a high strength, low swelling material capable of operation to a damage fluence of 3.6×10^{23} n/cm². Using doubling time and fuel cycle cost as the bases for judgment, an optimum pin outer diameter (o.d.) range of 0.37 to 0.40 in. was found for both the near-term and advanced structural materials. Because of practical limits of the Fast Test Reactor testing, an o.d. of 0.37 in. is recommended. The optimum pin diameter was found to be independent of fuel burnup. From a doubling time point of view, no incentive was found for development of carbide fuel with a burnup greater than 80 to 90 MWd/kg.

5351 DESIGN OF PROTOTYPE CARBIDE SUBASSEMBLIES AND AN EVALUATION OF PROOF-TESTING PLANS IN THE FFTF. Vittl, J.A.; Doherty, P.K.; di Lauro, G.F.; Gilbertson, J.C.; Stuteville, D.W. (Combust Eng Inc, Windsor, Conn). Nucl. Technol.; 26: No. 4, 442-451(Aug 1975).

Three subassembly configurations were designed, each capable of being directly substituted into FTR driver positions and made compatible with the FTR geometry, fuel handling, power, temperature, subassembly flow rates, and pin-bundle pressure drop. Two sodium-bonded designs, one with 91 fuel pins with a 0.370-in. o.d. and the other with 127 fuel pins with a 0.315-in. o.d. per subassembly were established. Calculated peak linear power and peak discharge burnup slightly exceed present commercial design objectives of 30 kW/ft and 73 MWd/kg. Individual assembly power history, flux, and enrichments are represented quite well in the FTR for commercial outer-zone fuel assemblies. Inner-zone operating conditions, however, are not simulated as closely. Impact effects of the proof-test assemblies on FTR operation are judged to be manageable.

5352 DESIGN AND MANUFACTURE OF GAS TAGS FOR FFTF FUEL AND CONTROL ASSEMBLIES. Strand, C.A.; Schenter, R.E. (Westinghouse Hanford Co, Richland, Wash). Nucl. Technol.; 26: No. 4, 472-479(Aug 1975).

The Fast Flux Test Facility being built near Richland, Washington for materials and component testing for development of the liquid-metal fast breeder reactor will employ "gas tagging" for locating failed fuel and control elements in the Fast Test Reactor (FTR). The fuel and control pins are "gas tagged" by loading a capsule with specially blended krypton and xenon isotopes. The encapsulation of the tag gas employs a unique application of laser technology: Stainless-steel capsules within a sealed glass-covered fixture containing the tag gas are first pierced and then seal welded with a laser beam. After inspection, the capsules are loaded in pins, and the gas is released by piercing the capsule with an electromagnetically activated internal penetrator. If the pin should develop a leak, the gas is released and the defect assembly is then located by mass spectrometric analysis of the reactor cover gas. Capsule filling yield for 40 000 capsules fabricated for 2 FTR cores was approx. 95%.

5353 EFFECT OF A LOSS OF FLOW ON A SINGLE LMFBR-TYPE FUEL PIN. Kwast, H. (React Cent Ned, Petten, Neth). Nucl. Technol.; 26: No.

4, 430-441(Aug 1975).

Capsule irradiations are being performed on fast reactor fuel pins in a sodium environment under loss-of-cooling conditions to determine the consequences of a fuel-pin failure. Special emphasis is given to the extent of possible fuel-sodium interactions (FSI) and the relocation of the fuel. The fuel pins were neither preirradiated nor prepressurized. So far no evidence has been found for a violent FSI on fuel ejection and sodium re-entry. Nevertheless, most probably a moderate FSI resulted from some fuel particles dropping into liquid sodium. These experiments indicate that considerable clad melting can occur before the fuel stack disintegrates. Considerable fuel relocation can occur, which can ultimately lead to failure of the sodium containment.

5354 EXTINGUISHING SYSTEM BY ISOLATION OF THE LIQUID FUEL. Colome, J.; Duchene, A.; Regnier, J. (to CEA, 75 - Paris (France)). French Patent 2,319,392/A/. 31 Jul 1975. 12p. (In French).

This invention concerns a system for extinguishing a sodium fire (or any other fuel in liquid sheet form) which complies with the safety requirements of a nuclear plant. To this effect, the system considered comprises a bottom receptacle for collecting the liquid fuel and an upper cover closing the receptacle. The cover has inclined flow drains for the liquid fuel and openings to allow the liquid through at the lower ends of these drains. These openings are closed by retractable shutters which withdraw under the pressure of the liquid in the drains and shut automatically after the liquid has passed through into the receptacle.

5355 SYSTEM FOR GUIDING A CONTROL ROD. Garchon, B.; Thaon, Y. (to CEA, 75 - Paris (France)). French Patent 2,319,958/A/. 31 Jul 1975. 7p. (In French).

The invention concerns an appliance to guide and maintain in a vertical position a control rod of a nuclear reactor and applies in particular to fast neutron nuclear reactors cooled by a liquid metal. An essential mandatory condition for the proper operation of such a control rod consists in efficient cooling of its absorbing part, through the correct flow of the liquid metal (generally sodium) cooling the core and flowing inside the rod sheath from the bottom upwards. The invention concerns a solution which, whilst ensuring that the rod will slide in its sheath without jamming, will leave the upper part of the sheath free.

5356 SYSTEM EXTINGUISHING A FIRE BY INSULATING A LIQUID FUEL. Colome, J.; Duchene, A.; Regnier, J. (to CEA, 75 - Paris (France)). French Patent 2,319,392/A/. 31 Jul 1975. 12p. (In French).

The invention refers to a system for quickly extinguishing a liquid fuel body on fire by insulating it completely from the ambient air. It applies particularly to the case of a high temperature liquid sodium sheet flowing accidentally from a circuit belonging to a fast neutron reactor. The system in question includes a lower receptacle for collecting the liquid fuel and a top cover shutting off the receptacle. This cover has inclined channels to take the liquid fuel flow and openings to allow this liquid through at the bottom end of the channels. These openings are closed by retractable shutters moving away under the pressure of the liquid in the channels and closing automatically after the liquid has

flowed into the receptacle .

5357 GUIDING SYSTEM FOR A CONTROL ROD.

Garchon, B.; Thaon, Y. (to CEA, 75 - Paris (France)). French Patent 2,319,958/A/. 31 Jul 1975. 7p. (In French).

This invention concerns a system for guiding and maintaining in a vertical position a control rod for a fast neutron nuclear reactor cooled by a liquid metal. These rods are fitted to slide in fixed sleeves, located at given places in the reactor core, and taking the shape of boxes with a regular polygonal straight cross section, open at their upper end and fitted with a positioning foot at their lower end. Efficient cooling of the absorbing part of the rod is assured through a proper flow of the liquid metal (sodium) cooling the core and flowing inside the control rod sleeve from the bottom upwards as from the bottom foot to the open top end. Most of this flow is forced to pass inside the rod itself and not between the rod and the sleeve in which it slides. The system, whilst ensuring that the rod slides in its sleeve without any risk of jamming, leaves the upper part of the sleeve free.

5358 TANK OF SODIUM COOLED FAST REACTOR.

Naito, K.; Ozawa, Y.; Minato, A.; Yokota, N.; Sato, Y. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-15980/A/. 30 Jul 1975. 4p. (In Japanese).

A tank is described which can safely and reliably accommodate high temperature sodium containing radioactive substance in case of occurrence of an accident in a sodium system and thus prevent spread of contamination. A sodium drain duct inserted into a tank from above the tank is provided at the position of its lower end with a buffer means for preventing direct flow-down of sodium to a bottom plate. A means for preventing the discharge of radioactive substance to the cover gas is provided above the lower end of the sodium drain tube so as to surround the sodium drain tube.

5359 FUEL ROD FOR FAST BREEDER. Miki, K. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-15,997/A/. 30 Jul 1975. 3p. (In Japanese).

The object of the invention is to improve heat-transfer factor of gap by reducing the quantity of loading fuel and alleviating the reduction of the ratio of presence of He gas in the gap due to combustion of fuel through reduction in the length of a blanket section in an axial direction. The fuel rod comprises a fuel pellet of a nuclear fission material consisting of Pu or U or a mixture thereof, a pellet consisting of beryllium oxide and a pellet of a parent material consisting of natural uranium or deteriorated uranium. Neutrons are decelerated by Be to increase the ratio of capture of U^{235} neutrons in the natural uranium or deteriorated uranium so as to increase the quantity of production of Pu^{239} . Further, with the He gas produced as the result of the nuclear reaction between Be and neutrons the ratio of presence of He gas in the gap is increased to permit alleviation of the reduction of the heat-transfer factor of the gap.

5360 INSULATING AND PROTECTING SYSTEMS FOR A CIRCUIT. Lemerrier, G. (to CEA, 75 - Paris (France)). French Patent 2,319,075/A/. 25 Jul 1975. 6p. (In French).

The invention concerns a device for

insulating and protecting systems or pipework carrying liquid sodium in fast neutron nuclear reactor installations or water or superheated steam. This device considerably lowers the risks whilst making it possible to give the insulating improved mechanical strength, without limiting its thermal protection performance and particularly to build into this insulating a protection and safety barrier against projections of the fluid outwards should the system burst accidentally. To this effect, the device considered includes on the outer surface of the system at least two successive windings of a continuous and long strip composed of a flat sock in knitted metal, comprising transversal openings to provide communication between the inside and outside of the sock, such openings allowing the insertion of thin successive metal sheets extending over the length of the strip.

5361 DEVICE OF THERMAL INSULATION AND MECHANICAL PROTECTION OF A CIRCUIT.

Lemerrier, G. (to CEA, 75 - Paris (France)). French Patent 2,319,075/A/. 25 Jul 1975. 6p. (In French).

The invention relates to circuits or conduits through which sodium flows (fast neutron nuclear reactor facilities) or else water or superheated steam. It concerns a lagging system which, without limiting the efficiency of the lagging in terms of thermal protection at high temperature, gives it improved mechanical strength and forms in this lagging a protective and safety barrier against projections of fluid outwards should the system burst accidentally. With this in view, the characteristic of the system under consideration comprises, on the outer surface of the system, at least two successive windings of a continuous long strip, composed of a flat metal mesh sock, fitted with transversal openings for communication between the inside and outside of the sock. These openings allow the introduction of successive thin metal sheets, extending along the length of the strip.

5362 INSULATING STRUCTURE. Lemerrier, G. (to CEA, 75 - Paris (France)). French Patent 2,319,073/A/. 24 Jul 1975. 7p. (In French).

The invention concerns an insulating structure for protecting a wall against a high temperature atmosphere. This structure has the advantage of being adaptable to the shape of the area to be protected, particularly in an annular space. It applies in particular where the wall to be protected is in an aerosol or liquid metal vapour atmosphere (sodium in particular) inside a fast neutron reactor vessel. To this end, the structure considered is built of numerous components joined together, each formed of two long angle bars, inter-slotted with reduced play between their parallel faces, these two angle bars themselves enclosing a soft metallic filling. The soft filling is formed of a stack of metal cloth, netting or knitting or stuffed with wool or chippings.

5363 THERMAL INSULATION SYSTEM FOR A VESSEL. Lemerrier, G. (to CEA, 75 - Paris (France)). French Patent 2,319,074/A/. 24 Jul 1975. 13p. (In French).

This invention concerns a thermal insulation system for the vessel wall of a nuclear reactor cooled by water, steam, gas or liquid metal. It applies in particular to the main vessel of a fast neutron nuclear reactor. This system includes numerous superimposed elastic lagging

elements, fitted in the annular space between the main vessel and the inner vessel. These elements extend circumferentially around the common centre of both vessels.

- 5364 FAST REACTOR. Ilyunin, V.G.; Murogov, V.M.; Shmelev, A.N. Czech Patent 158,840/B/. 15 Jul 1975. 17p. (In Czech).

A fast reactor is designed and schematically shown which has a configuration of fuel elements in the core and a heat removal system which permit increasing the energy load of the core without considerably increasing the core size. Several methods of arranging the primary and the auxiliary fuel elements in the reactor core are described. The energy load of the core may be increased by a factor of 1.3 to 1.5. A table is given comparing the parameters of the proposed reactor with those of known reactors.

- 5365 COOLING SYSTEM FOR THE HIGH WALL OF A NUCLEAR REACTOR VESSEL. Failla, H.; Schaller, K.; Vidard, M. (to CEA, 75 - Paris (France)). French Patent 2,316,705/E/. 2 Jul 1975. 5p. (In French).

Addition to the French patent document 7429648.

The invention concerns a cooling system for the high wall of a nuclear reactor vessel, particularly the main vessel of a liquid metal cooled fast neutron reactor. This system is composed of coils carrying a coolant which are in contact with an intermediate heat conducting substance in contact with the high wall. This intermediate substance can be a solid or gas. Among the appropriate solids mention can be made of thermal concrete and metals such as aluminum, copper or their alloys used in mass form or divided up (balls for instance). The gas that may be used are helium or nitrogen for instance. Should a gas be employed it is contained in a leak proof jacket. When the substance is a solid, all that is required is a bottom bearing surface.

- 5366 FUEL ASSEMBLIES FOR NUCLEAR REACTOR. Marmonier, P.; Mesnage, B.; Nervi, J.C. (to CEA, 75 - Paris (France)). French Patent 2,316,703/A/. 2 Jul 1975. 15p. (In French).

The invention concerns an assembly for a sodium cooled fast neutron reactor. The invention aims to simplify the assembly, whilst also assuring a reliable and uniform flow rate for each type of assembly entering into the construction of the core. The invention further aims to prevent any assembly, once ready to be set in position in the core, from being inserted by mistake in a place different from that assigned to it. An open casing contains the bundle of clad pins filled with a fuel. This casing is connected to a foot composed of a hollow cylindrical body, closed at its lower end by a plug. This cylindrical body is designed to be inserted in a vertical position in a hollow housing belonging to the diagrid. Inside the foot there is an appliance for pre-setting the flow rate of the coolant.

- 5367 IMPROVEMENT TO HIGH TEMPERATURE ALLOYS. Bleiberg, M.L.; Diamond, S.; Rowcliffe, A.F.; Spitznagel, J.A. (to Westinghouse Electric Corp., Pittsburgh, PA (USA)). French Patent 2,330,776/A/. 1 Jul 1975. 14p. (In French).

The description is given of an alloy exhibiting good physical properties at high temperatures up to 1325°F (720°C), low corrosion in molten sodium and low swelling under irradiation. The alloys consists

essentially of 14 to 19% chromium, 25 to 35% nickel, 2 to 3% molybdenum, 0.1 to 1% silicon, up to 0.5% manganese, 0.03 to 0.05% carbon, up to 0.01% sulphur, up to 0.01% phosphorus, up to 0.01% boron, up to 0.01% oxygen, up to 0.02% nitrate, small amounts of accidental impurities, the balance being iron.

- 5368 SODIUM CIRCULATING SYSTEM FOR A REACTOR. Sasazawa, A.; Nakao, N. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1977-4,991/A/. 30 Jun 1975. 4p. (In Japanese).

A method is described to provide an overflow system in a mechanical sodium pump, which eliminates the provision of an overflow column and makes it easy to install a guard vessel and minimizes entry of gas into sodium by passing it through an overflow tank. Structure: Overflowed sodium in a circulating pump is once introduced into an overflowing tank in an overflowing system provided in order to maintain a highest liquid level within a reactor vessel constant, and the thus introduced sodium is pumped into the reactor vessel by a pumping electromagnetic pump thereby providing positive treatment without using the overflow column and without possibility of entry of gas.

- 5369 SYSTEM FOR FIXING INSULATING PANELS. Aubert, M.; Lemerclier, G. (to CEA, 75 - Paris (France)). French Patent 2,315,055/A/. 16 Jun 1975. 8p. (In French).

This invention concerns a system for supporting and hanging insulating panels on a wall. It particularly applies to securing panels forming the protective lining of the horizontal sole and sides of the upper shield of the vessel and jacket of a fast neutron nuclear reactor (or a protective lining for the vertical rings around the component tubes entering into the design of such a reactor). An improvement enables the panels to be hung and fixed with the added advantage of allowing a slight movement of these panels with two degrees of play with respect to the plane of the wall requiring protection.

- 5370 LIKE-PROOF INSULATING PANEL. Lemerclier, G. (to CEA, 75 - Paris (France)). French Patent 2,315,054/A/. 16 Jun 1975. 12p. (In French).

The invention concerns a leak-proof insulating panel, designed for the operating conditions of an LMFBR reactor, enabling the structures of the reactor to be protected against the effects of temperature and those of corrosion by the liquid metal. The panel in question forms a flattened leak-proof containment, suitable for application against the wall needing protection. The containment is delimited by two parallel sealing metal plates welded together peripherally and providing between them a closed space stacked with wire cloth or netting and perforated plates or grids, the empty spaces between the cloth and perforated plates being filled with a neutral gas.

- 5371 DEVICE FOR PROTECTING DEFORMATIONS OF REACTOR CORES. Kato, Y.; Urushihara, H. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1976-148198/A/. 13 Jun 1975. 5p. (In Japanese).

The object of the invention is to provide a fluid pressure cylinder which is operated according to change in temperature of coolant for a reactor to restrain or release a core, to simply and effectively protect deformation of the core. A closed fluid pressure cylinder

interiorly filled with suitable fluid is disposed in peripherally equally spaced relation in an annular space between a core barrel of a reactor and a reactor vessel. A piston is mounted in fluid-tight fashion in a plurality of piston openings made in the cylinder, the piston being slidably moved according to expansion and contraction of the fluid filled in the cylinder. The piston has a movable frame mounted at the foremost end thereof, the movable frame being moved integral with the piston, and the surface opposite the mount thereof biasing the outermost peripheral surface of the core.

- 5372 PRIMARY COOLING DEVICE FOR A SODIUM COOLED FAST BREEDER. Kawashima, H. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1976-148199/A/. 13 Jun 1975. 4p. (In Japanese).

The object of the invention is to positively position a return pipe nozzle for leaked sodium in pump to a given depth below sodium level thereby eliminating possible entry of gas and the provision of a pump overflow column. In a primary cooling device for a sodium cooled fast breeder comprising a reactor vessel, an intermediate heat exchanger, a cold leg mechanical circulating pump, a pipe for leaked sodium in pump, and a guard vessel, a return pipe is provided to directly return leaked sodium in the circulating pump to a medium pressure and low temperature portion of the reactor vessel. Since the returning destination of the return pipe for leaked sodium in the pump is within the reactor vessel, the liquid level of the circulating pump is positively maintained above the liquid level of the reactor vessel so that the return pipe nozzle for leaked sodium in pump is positively positioned to a given depth below sodium level.

- 5373 THREE-COMPONENT HEAT EXCHANGER. Vinzens, K.; Berner, F. (to Internationale Atomreaktorbau G.m.b.H. (INTERATOM), Bergisch Gladbach (Germany, F.R.)). German (FRG) Patent 2,360,257/A/. 5 Jun 1975. 15p. (In German).

A three-component heat exchanger is proposed in which two media are carried in separate tube systems designed as approximately vertical, concentric, cylindrical (or plane) tube walls arranged in a vessel filled with a stagnant fluid compatible with the two media in such a way that a small clearance is left between the two tube walls. Such heat exchangers can be employed in nuclear power plants equipped with circuits carrying sodium and water (vapor). If the two media have very different enthalpies, the cross section or the arrangement of the different tubes relative to each other is balanced out. A lead-bismuth eutectic is proposed as the contact medium.

- 5374 NUCLEAR REACTOR FUEL ASSEMBLY. Marmonier, P.; Mesnage, B.; Nervi, J.C. (to CEA, 75 - Paris (France)). French Patent 2,313,742/A/. 5 Jun 1975. 7p. (In French).

This invention refers to fuel assemblies for a liquid metal cooled fast neutron reactor. Each assembly is composed of a hollow vertical casing, of regular polygonal section, containing a bundle of clad pins filled with a fissile or fertile substance. The casing is open at its upper end and has a cylindrical foot at its lower end for positioning the assembly in a housing provided in the horizontal diagrid, on which the core assembly rests. A set of flat bars located on the external surface of the casing enables it to be

correctly orientated in its housing among the other core assemblies.

- 5375 METHOD OF PREVENTING SODIUM FROM FLOWING WHEN PIPES OF A FAST BREEDER REACTOR ARE INJURED. Nakai, Y.; Yamagishi, Y.; Koga, T. (to Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Japanese Patent 1976-141,992/A/. 2 Jun 1975. 3p. (In Japanese).

When the core cooling pipe on the inlet side is injured and as a consequence the pressure gage detects an abnormal condition of fluid pressure in the inlet nozzle, the valve is opened to allow high pressure sodium to inject into the inlet nozzle through a high pressure sodium supply pipe, thereby blocking a back-flow of sodium in the high pressure chamber into the core cooling pipe.

- 5376 BUBBLE SEPARATOR FOR FAST BREEDER REACTORS. Nakai, Y.; Yamagishi, Y.; Koga, T. (to Power Reactor and Nuclear Fuel Development Corp., Tokyo (Japan)). Japanese Patent 1976-141,993/A/. 2 Jun 1975. 3p. (In Japanese).

This separator comprises an up-pipe which extends through a core support plate positioned at the upper part of a high pressure chamber in a reactor vessel, a flowmeter and a flow regulating valve mounted on the upper end of the up-pipe, and a down-pipe mounted at the outlet side of the flow regulating valve to return fluids into the vessel, whereby when the flowmeter detects a flow passage of bubbles, the flow regulating valve is fully opened to remove gases from the high pressure chamber.

- 5377 CONTROL ROD ASSEMBLY FOR NUCLEAR REACTOR. Garchon, B.; Marmonier, P.; Thaon, Y. (to CEA, 75 - Paris (France)). French Patent 2,313,743/E/. 2 Jun 1975. 6p. (In French).

Addition to the French Patent document 7046296.

In the main patent a description was given of a control rod assembly for a fast neutron, liquid metal cooled, power reactor. This assembly includes a fixed sheath and a control rod with an absorbant part, a head enabling the rod to be held and moved vertically from a low position where the absorbant part is in the core and a high position where it is out of the core, and a foot on a level with the core when the rod is in the high position. Above and below the core, the sheath has inlets and outlets for the coolant. A guide bearing is located between the rod and the sheath (above core level), at the upper part of the sheath and provides just the right amount of play for the flow rate of the coolant circulating between the sheath and the rod to be an insignificant fraction of the amount flowing in and out of the sheath. This invention refers to an improvement which, whilst retaining the advantages proper to the operation of the guide bearing in question, allows a slight lack of alignment between the axis of the sheath and that of the control rod mechanism which is carried by a revolving plug placed inside the slab forming the upper closure of the reactor.

- 5378 METHOD FOR DRAINING LIQUID METAL. Yokota, N.; Naito, K.; Ozawa, Y.; Minato, A.; Fukushima, Y. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1976-137088/A/. 23 May 1975. 7p. (In Japanese).

The object of the invention is to connect a pit chamber, a primary cooling system piping and a machine room to a damp tank through a

drain passage thereby recovering leaked liquid metal to prevent the surrounding atmosphere from being contaminated. The pit chamber, primary cooling system piping and primary cooling system machine room are connected to a damp tank through a drain passage. A plug made of metal (such as lead, tin, etc.) having a low melting point is mounted at a connection between the aforementioned chambers and the drain passage. When high temperature liquid metal (Na, K, Li, etc.) are stayed on the plug, the plug is fused by heat to open the passage. Also, a gas take-out passage for allowing vapor from the damp tank to be escaped is disposed around the drain passage so that fluid flowing through the drain passage and fluid flowing through the gas take-out passage may be heat-exchanged. The materials suspended in the gases may be removed by a vapor trap or the like.

- 5379 APPARATUS FOR CONTROLLING A CORE COOLING SYSTEM IN CASE OF EMERGENCY. Watanabe, T. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1976-136096/A/. 19 May 1975. 4p. (In Japanese).

An apparatus is disclosed whose purpose is to securely perform low level control of main cooling system, particularly, vapor generator at the time of minor trouble such as power failure in a fast breeder plant and to securely and safely perform core cooling in combination with auxiliary cooling system. When power failure occurs, a sequence control gives instructions to start the auxiliary system secondary pump and blower and sodium temperature at outlet of air cooler is controlled constant by a damper control. Primary and secondary pumps in the main cooling system are tripped by respective controls and shifting into pony motor drive. By a pressure control, a superheater separating valve is closed and a flush valve is opened so that a vapor generator is formed into individual operation of evaporator and is switched to a low level control system. The water pump is shifted to motor drive and a water supply controller keeps a temperature on low temperature side of the main cooling system constant.

- 5380 METHOD OF REMOVING OXYGEN FROM MOLTEN METALS IN NUCLEAR TECHNOLOGY. Jakes, D. Czech Patent 158,060/B/. 15 May 1975. 2p. (In Czech).

The method is claimed of the electrolytic removal of oxygen from molten metals in nuclear technology, in which ionized oxygen atoms are converted by a system blocking the electron beam. This blocking system consists of one or more alkaline earth metal oxides and of a solid electrolyte. The formed gas molecules are removed by a stream of inert or reduction gas characteristic by the fact that the ionized oxygen atoms are converted by a system blocking the electron beam at a potential stability of maximally 10 mV/min, or to advantage at a stability of maximally 5 mV/min. At this stability value considerably less damage occurs to the electrolytic tube as a result of corrosion and solid electrolyte life is considerably longer.

- 5381 ROTARY SHIELD PLUG. Ugawa, Y. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1976-130,790/A/. 12 May 1975. 3p. (In Japanese).

When the cover gas pressure abnormally increases with the seal alloy in the fused state, the level of the seal alloy within an annular space on the outer side of the liquid

seal is raised. With the rise of the level of the seal alloy a float ring is also raised under the guide of a ring holder rod. When the float ring is raised up to an upper check plate, a borderline gap formed in the upper check plate is blocked thereby to check a further rise in the liquid and prevent gas leakage.

- 5382 FUEL ASSEMBLY. Tsutsumi, K.; Kato, Y.; Aoi, H. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1976-130,786/A/. 9 May 1975. 5p. (In Japanese).

An elongated fuel pin is provided within a cylindrical member, and a seal member consisting of beryllium or an alloy thereof which is provided as a neutron decelerating material layer below the pin. Further, the seal member is covered with a stainless steel cover for reinforcing the fragility of beryllium.

- 5383 VALVES FOR FAST SODIUM REACTORS. Vinci, M. Quad. Pignone; No. 20, 37-42(May 1975).

The problems of valving the liquid sodium used for cooling fast nuclear reactors are discussed. The standard specifications are given. Methods of manufacturing valves to meet these specifications are described, and examples of specific developmental designs and methods of test are presented. What is described represents the present state of technology used by Nuovo Pignone, Florence, Italy. Research must now proceed on valves suitable for industrial-scale power plants, such as 40-inch valves as compared to the 1- to 8-inch valves so far manufactured.

- 5384 VAPOR GENERATOR SYSTEM EQUIPMENT FOR FAST BREEDER REACTOR. Kawakami, H.; Mori, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1976-127,993/A/. 30 Apr 1975. 4p. (In Japanese).

At the occurrence of a sodium-water reaction accident, cover gas pressure in an evaporator is adapted to suddenly increase due to gas generation to cause rupture of a rupture disc, whereby the gas generated is discharged through a gas discharge duct, a container and a cyclone separator to the atmosphere. At the instant that the reaction ends, a drain valve and a pressure release valve are opened, whereby Na in the evaporator falls by its own weight down to a damp tank to cause the cover gas within the damp tank to be supplied through a cover gas duct, a container, and a gas discharge duct to the evaporator.

- 5385 THERMAL PROTECTION DEVICE FOR A NUCLEAR REACTOR SLAB. Boy-Marcotte, J.-L. (Societe Bertin et Compagnie, 78 - Plaisir (France)); Chevalier, G.; Gerij, J.; Leheu, J. (Societe Technique pour l'Energie Atomique (Technicatome), Centre d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)). (to CEA, 75 - Paris (France)). French Patent 2,309,957/A/. 30 Apr 1975. 9p. (In French).

Description is given of a system for the thermal protection of the closure slab of a reactor vessel with respect to a gaseous medium topping the primary fluid of the vessel, passages being made in the slab and its walls in contact with the gaseous medium being covered by a metal plate, called a shield rim. The system comprises a relatively thin annular metal plate that is placed horizontally under the slab and is provided with holes facing the passages and marks with the shield rim plate

enclosed spaces confining an inert gas. The inside edge of the plate and the free edges delimited in the latter by the holes are connected to the edging sheet metal of the slab through a leaktight collar and the outer edge of the plate is connected by large radius connections to the sheet metal rim of the shield.

- 5386 METHOD OF CONTROLLED MAINTENANCE OF REACTOR. Orit, S.c.; Yoshizawa, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1976-124792/A/. 22 Apr 1975. 3p. (In Japanese).

A method is disclosed to improve the operability of maintenance of a reactor at the time of function of a seal section for sealing radioactive material by employing a filler pouring method. Structure: When a rupture takes place in a seal bellows provided in a high-speed breeder reactor control rod drive mechanism upper guide duct, a sealing compound is poured from a back-up seal gas inlet hole formed in a control rod drive mechanism housing mounted on a lift rod mechanism and agglomerated to seal the upper portion of the guide duct so as to maintain gas-tightness on the inside, thereby preventing loss of reactor cover gas due to leakage.

- 5387 FACILITY OF AEROSOL FILTRATION. Duverger de Cuy, G.; Regnier, J. (to CEA, 75 - Paris (France)). French Patent 2,307,566/A/. 18 Apr 1975. 6p. (In French).

Said invention relates to a facility of aerosol filtration, particularly of sodium aerosols. Said facility is of special interest for fast reactors where sodium fires involve the possibility of high concentrations of sodium aerosols which soon clog up conventional filters. The facility intended for continuous operation, includes at the pre-filtering stage, means for increasing the size of the aerosol particles and separating clustered particles (cyclone separator).

- 5388 DEVICE OF IRRADIATION AND MEASUREMENT OF FUEL ELEMENTS FOR NUCLEAR REACTORS. Descamps, J.M. (to CEA, 75 - Paris (France)). French Patent 2,308,170/A/. 18 Apr 1975. 9p. (In French).

The description is given of a device for irradiating a cylindrical fuel element (and more particularly the fuel element of a sodium cooled fast reactor) which enables the behavior of such an element to be tested in a pool reactor. This system includes a closed containment extended at its upper part by a tube, the containment and tube being filled with a fluid identical to the coolant of the reactor. The containment able to take the element and the tube are greater in diameter than the element. The gear for moving the element from the containment to the upper end of the tube and the facilities for measuring the diameter of the element when it is engaged in the tube are located between the containment and the upper end of the tube. Means are also provided to maintain the fluid at a temperature significantly equal to that it would have in the reactor.

- 5389 NUCLEAR REACTOR ASSEMBLY. Garchon, B.; Lions, N.; Marmonier, P.; Mesnage, B.; Nervi, J.C. (to CEA, 75 - Paris (France)). French Patent 2,306,507/A/. 3 Apr 1975. 8p. (In French).

Improvements brought to the assemblies of sodium cooled fast reactors of the integrated type are presented. According to the patent,

all the assemblies have a standard head, whatever type of pins they contain, in order to make their handling easy. This head is fitted on the upper part and in axial prolongation of a long hollow casing of polygonal cross section constituting the assembly body. This head is provided with a cylindrical sleeve that has a reflecting power constant towards ultrasonic radiations in order to allow the accurate localization of each assembly.

- 5390 LIQUID LEVEL CONTROL SYSTEM OF FAST REACTOR SECONDARY COOLING SYSTEM. Mori, K.; Matsumoto, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1976-113094/A/. 31 Mar 1975. 8p. (In Japanese).

A system is disclosed whose object is to minimize the range of the liquid level variation of the cooling system and reduce the time required for the liquid level control by sealing the gas of a cover gas respiration system which acts upon an evaporator and pump overflow column. In liquid level control by the cover gas pressure of a high-speed reactor secondary cooling system, upon occurrence of a sudden change in the rate of flow of the recirculated liquid, automatic check valves provided in an evaporator and pump overflow column cover gas respiration system are completely or substantially closed, while at the same time the recirculation cooling medium is sucked up and an automatic check valve provided in the overflow system is closed.

- 5391 ATOMIC POWER PLANT. Kawakami, H. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1976-113096/A/. 31 Mar 1975. 6p. (In Japanese).

The object of the invention is to permit decay heat to be reliably removed after reactor shut-down at such instance as occurrence of loss of power by means of an emergency water supply pump. An atomic power plant has a closed cycle constructed by connecting a vapor generator, a vapor valve, a turbine having a generator, a condenser, and a water supply pump in the mentioned order, and is provided with an emergency water supply pump operated when there is a loss of power to the water supply pump, a degasifier pressure holding means for holding the pressure of the degasifier by introducing part of the vapor produced from said vapor generator, and a valve for discharge to atmosphere provided on the downstream side of said vapor generator.

- 5392 CONTROL SYSTEM FOR EMERGENCY CORE COOLING SYSTEM. Mae, Y. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1976-111596/A/. 26 Mar 1975. 4p. (In Japanese).

A system is disclosed whose object is to reduce thermal stress produced when the apparatus and ducts constituting the reactor and cooling system are rapidly cooled down at the time of operation of an LMFB emergency core cooling system. Structure: Decay heat produced from the reactor is led through a primary cooling system and an intermediate heat exchanger to an air cooler for removal, and a rise in reactor output temperature due to decay heat is detected by a detector and compared in a comparator with a signal from a secondary cooling system flow rate detector. The pump of the emergency cooling system is controlled through a PI controller and pump output controller such that heat in a quantity corresponding to the decay heat is supplied to the air cooler according to the difference signal from the comparator. A damper is

controlled according to the difference in the signal from a cooler output temperature detector from a preset value and also a difference in the signal from the cooler inlet temperature detector and difference of the signal from a bias setter, whereby the quantity of cooling air supplied to the air cooler is controlled.

- 5393 DOOR VALVE. Honda, N.; Sakaguchi, S.; Goto, T.; Horikawa, T. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1976-110194/A/ 24 Mar 1975. 3p. (In Japanese).

Leakage of inner fluid in a door valve is blocked by providing a bellows mechanism in a portion where a valve rod penetrates an oil hydraulic cylinder and by sealing opposite side valve seats of the valve body with an oil hydraulic type seal mechanism. A door valve comprising a valve body movably provided within a valve casing, an oil hydraulic mechanism driving a valve rod directly connected to the valve body, a bellows mechanism provided on a portion where a valve rod penetrates the cylinder of the oil hydraulic mechanism and an oil hydraulic seal mechanism provided on the inner side of an inlet of the valve casing opposing a valve seat of the valve body, thereby isolating such components as the reactor container, casker, relay container and fuel storage tank at the time of replacement of fuel and reliably preventing leakage of liquid metal vapor containing radioactive material.

- 5394 ROTARY PLUG. Orit, S.; Fujiki, K. (to Tokyo Shibaura Electric Co. Ltd., Kawasaki, Kanagawa (Japan)). Japanese Patent 1976-108194/A/. 19 Mar 1975. 3p. (In Japanese).

A rotary plug is provided for a reactor comprising a large disc and a small disc disposed eccentrically thereto and rotatably using liquid metal as the coolant, the large and small discs having sliding surfaces formed with stepped portions in which a resilient member is interposed, thus minimizing the amount of solidified and deposited Na steam to effect smooth rotation. A V-shaped annular spring having a surface hardened layer placed in intimate contact with a large rotary plug and a small rotary plug is provided to block, during the operation of the reactor, upward movement of Na steam which is produced from Na liquid surface and tends to be moved up after entry into an annular space, and to, when the furnace is stopped in operation to rotate the rotary plugs, minimize a shearing stress created when the small rotary plug is lifted and to minimize a reverse rotational torque created when the large and small rotary plugs are rotated relative to each other.

- 5395 METHOD OF CONTROLLING STEAM TEMPERATURE OF A FLUID HEATING SEPARATION TYPE STEAM GENERATOR. Iwashita, T.; Monta, K. (to Nippon Atomic Industry Group Co. Ltd., Tokyo). Japanese Patent 1976-108196/A/. 19 Mar 1975. 5p. (In Japanese).

A system is disclosed whose object is to keep constant the stability and normal deviation in the entire control system by connecting an element of variable gain substantially in proportion to a preset load in series with the ordinary PID type control system. Changes in steam temperature at an evaporator outlet due to changes in sodium flow rate are detected by a thermocouple. The resultant detection signal is compared with a preset value of the steam generator output temperature, and a portion proportional to the difference between them is added as an

operating signal, the operating signal also being used as a sodium flow rate control signal coupled to a sodium flow rate control means. In this method of control of vapor temperature of a fluid heating separation type steam generator, a control gain variable means is connected in series with a temperature control system to obtain control substantially proportional to the preset load.

- 5396 SAFETY VALVE SYSTEM FOR A NUCLEAR VESSEL COVER GAS. Itami, K. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1976-104190/A/ 12 Mar 1975. 4p. (In Japanese).

Leakage of a radioactive gas from a valve seat is presented by providing a burst plate downstream of a pressure relief valve. A pressure relief valve is disposed at the side of a nuclear reactor. A first burst plate is provided at an outlet opening in the valve in order to prevent the leakage of a radioactive gas. A compression system is provided to break the burst plate. A second burst plate is provided in the compression system thereby to bring the pressure in the space surrounded by the pressure relief valve and the first and second burst plates into equality with the pressure of the ambient atmosphere.

- 5397 PROTECTION DEVICE FOR REACTOR SAFETY. Miki, K.; Inoue, K. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1976-104189/A/. 12 Mar 1975. 3p. (In Japanese).

A molten fuel dropping from a reactor core is received in a container. The container is formed as the heater section of a heat pipe and is adapted to heat and gasify a medium of Li, Na, K or NaK in the pipe. The gasified medium is condensed by a radiator system and returned by capillary action to the heater (the container). An inert gas is ordinarily enclosed in the heat pipe in order to prevent the evaporation of the medium while the gas is exhausted from the pipe during an accident thereby to actuate the heat pipe.

- 5398 FUEL ASSEMBLY FOR A NUCLEAR REACTOR. Takeda, S.; Sakagami, M. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1976-104186/A/. 10 Mar 1975. 2p. (In Japanese).

To prevent the heat-induced bending of a fuel assembly for use in a fast breeder reactor due to a thermal difference radially of the fuel assembly within the reactor core. A wrapper tube is provided which has a corrugated or wavy wall portion on a higher output (or higher temperature) side of a reactor core thereby to absorb the elongation of the wall of the tube on the higher temperature side and to prevent the bending of the tube.

- 5399 FAILED FUEL DETECTOR FOR NUCLEAR REACTORS. Takahara, K. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1976-101693/A/. 4 Mar 1975. 4p. (In Japanese).

Cover gas is circulated by the suction of an ejector utilizing the potential energy of overflow sodium, thereby dispensing with power for cover gas circulation. Using the suction provided by the ejector the cover gas within the reactor container is withdrawn successively through a duct, vapor trap, failed fuel detector, and another duct. It is then mixed with sodium flowing through the overflow duct, and the mixture is led to an overflow tank for separation of gas and liquid. Next, the gas is returned to the reactor container through a duct, cover gas header and another duct.

5400 PRINCIPAL RESULTS OF OPERATION OF THE BOR-60 FACILITY. Kazachkovskii, O.D.; Afanas'ev, V.A.; Gryazev, V.M.; Kevrolev, V.P.; Kondrat'ev, V.I.; Nechaev, V.N.; Smirnov, A.M. *Sov. At. Energy (Engl. Transl.)*; 38: No. 3, 169-172 (Mar 1975).

General operating characteristics of the BOR-60 facility, fuel element tests, tests of the main equipment and additional measures for ensuring safety are described.

5401 EMERGENCY COOLING SYSTEM OF A NUCLEAR REACTOR. Ventre, E. (to Electricite de France, 75 - Paris). French Patent 2,330,123/E/. 18 Feb 1975. 6p. (In French).

This invention concerns a nuclear reactor emergency cooling system where the reactor core is composed of separate N modules, each module being cooled by a coolant kept flowing by a pump assigned to each module. This system applies in particular to liquid sodium cooled fast neutron reactors.

5402 ROTARY SHIELD PLUG. Maruya, K.; Okuma, T. (to Mitsubishi Atomic Power Industries, Inc., Tokyo (Japan)). Japanese Patent 1976-92994/A/. 12 Feb 1975. 4p. (In Japanese).

A device to shield Na vapor of a liquid metal coolant accommodated within the container of a fast reactor with a doublewall shield structure, thereby increasing the shielding reliability is described. A gap between the container wall of fast reactor container and rotary plug for closing the top opening of the reactor container is hermetically shielded at the upper end and is doubly shielded at the lower end by two shielding means. Of these shielding means, the first one projects from the container wall and is in contact with the bottom of the rotary plug, while the second shield means consists of a first annular member extending from the bottom of the rotary plug and a second annular member extending from the container wall and abutting the first annular member to define a space between the first and second annular members. Further, an inert gas circulating means is provided between a blow-in inlet positioned above the first shield means on the container wall and an exhaust outlet positioned above the space within the container wall.

5403 PROCESS OF FLATTENING OF THE HEAT RELEASE CURVE OF A NUCLEAR REACTOR CORE AND REACTOR CORE USING SAID PROCESS. Cachera, P. (to Electricite de France, 75 - Paris). French Patent 2,300,398/A/. 16 Feb 1975. 13p. (In French).

A procedure for flattening the heat release curve of the fuel elements in a fast reactor core is presented. Said core is made of vertical fuel elements with an axial channel large enough in diameter to make it possible for the molten fissile material to flow, in the event of a power excursion. Said axial channel diameter is reduced when going from the elements in the central core to the peripheral elements. Said decrease in diameter is proportional to the decrease in the neutron flux density, so as the conductivity integral to be maintained nearly constant.

5404 CONNECTION DEVICE OF AN APPARATUS IN A SYSTEM WITH STRUCTURES SUBJECT TO RELATIVE MOTIONS. Barneoud, R.E.; Guer, M. (Societe Nationale d'Etude et de Construction de Moteurs d'Aviation (SNECMA), 92 - Suresnes (France). Div. Hispano-Suiza); Roumailhac, J.; Lecouvreur, P. (Societe Technique pour l'Energie Atomique (Technicatome), Centre

d'Etudes Nucleaires de Saclay, 91 - Gif-sur-Yvette (France)). (to CEA, 75 - Paris (France)). French Patent 2,300,253/A/. 6 Feb 1975. 14p. (In French).

The invention relates to a system of sealed connection intended for sodium pumps of LMFBR type reactors. The fastening system is characterized in that the pump immersed in the liquid metal has its upper part suspended to the closure slab of the reactor shell through a first knee-joint system, and its lower part fastened by means of a second swivel-joint to a steady frame rigidly connected to the reactor vessel.

5405 SYSTEM OF BRACING INTENDED FOR A BUNDLE OF PARALLEL PINS INSIDE A FUEL ASSEMBLY OF NUCLEAR REACTOR. Marmonier, P.; Message, B.; Skoko, J.; Vayra, J. (to CEA, 75 - Paris (France)). French Patent 2,298,162/A/. 20 Jan 1975. 12p. (In French).

Description is given of a bracing system of the parallel pins of a bundle mounted inside a vertical axis open casing provided with a part supporting the pins so as to form a fuel or fertile assembly for a nuclear reactor. The plain structure endures quite well the effects of differential swelling of the pins, without entailing any pressure drop. Besides, the system is suitable in that a better stirring of the sodium flow through each fuel assembly is ensured while the volume percent of fuel inside the assembly is only weakly reduced.

5406 TEMPERATURE MEASUREMENT DEVICE. Fournier, C.; Lions, N. (to CEA, 75 - Paris (France)). French Patent 2,297,409/A/. 13 Jan 1975. 9p. (In French).

The present invention relates to a temperature measuring system that can be applied in particular to monitoring the temperature of the cooling liquid metal of the outlet of the core assemblies of a fast reactor. Said device combines a long hollow metallic pole, at least partially dipped into the liquid metal and constituting a first thermocouple junction between said pole, and two metallic conductors of different nature, joined at one of their ends to constitute the second thermocouple junction. Said conductors suitably insulated are arranged inside a sheath. Said sheath made of the same metals as the pole extends inside the latter and is connected with the pole through a soldered joint. Said reliable system permits an instantaneous measurement of a quantity representing the variations in the recorded temperature and a measurement of the mean surrounding temperature that can be directly used as a reference for calibrating the first one.

5407 PROTOTYPE FAST REACTOR DOUNREAY: SOME INTERESTING ASPECTS OF ITS DESIGN AND CONSTRUCTION. Outram, W.J. (UK At Energy Auth, Warrington, Lancs, Engl). *Proc. Inst. Electr. Eng. (London)*; 122: No. 1, 49-54 (Jan 1975).

Some of the more interesting aspects of the design and construction of the prototype fast reactor at Dounreay are discussed. These include the bringing together of the primary-vessel components, all of which are supported from the reactor roof, the construction of which involved the use of flotation. A description is included of the rotating shield and its function related to the charge procedures of the reactor. The use of the irradiated-cave facilities is described along with some of the special machines within this complex. 4 refs.

- 5408 TRANSIENT TEMPERATURE DISTRIBUTION IN REACTOR COOLANT CHANNEL FOLLOWING LOSS OF FLOW. Ozgu, M.; Chen, J.C. (Bethlehem Steel Corp, Pa). Lett. Heat Mass Transfer; 2: No. 1, 27-36 (Jan-Feb 1975).

Analytical solutions to the transient energy equation in a coolant filled channel subsequent to complete flow stoppage were obtained for different end boundary conditions and wall heat fluxes. Sinusoidal and stepwise heat flux distributions along the channel, and constant temperature and zero heat flux end boundary conditions were used. Analytical solutions for variations of temperature with time and axial position were obtained. Sample results were determined for liquid sodium with various values of initial Peclet number. 7 refs.

- 5409 LMFBR DESIGN BASIS ACCIDENTS AND THEIR ACCOMMODATION. Graham, J. pp 284-291 of Nuclear safety, 1975. Hetrick, D.L. (ed.). Hinsdale, IL; American Nuclear Society (1975). From ANS topical meeting on nuclear safety; Tucson, AZ, USA (5 Oct 1975). See CONF-751044--.

Accidents are defined of various consequences and probability for use as design bases in the design of reactor safety systems. In the past a core disruption was used to set requirements for containment barriers of small LMFBRs. It now seems that the use of such a design basis accident has created a misplaced emphasis. Greater safety is to be provided by the provision of reliability assurance, that such an accident is of such low probability that it is inappropriate as a design basis accident, and greater emphasis should be placed on the basic protection against accidents rather than on the mitigation of presumed consequences.

- 5410 CASE HISTORY OF FABRICATING NUMERICAL-CONTROLLED MACHINERY FOR THE NUCLEAR INDUSTRY. Lewis, J.F. Dearborn, MI; Society of Manufacturing Engineers (1975). 15p.

Working to tight tolerances, attaining specific mechanical properties and conforming to rigid specifications are some of the problems that occur in fabricating machinery for the nuclear industry. The equipment being built is the In-Vessel Handling Machine, which is to be used in the Fast Flux Test Facility. The largest fabrication for the machine is the Reactor Refueling Plug (RRP). The body of the RRP is a low-alloy steel forging (ASME SA508) that is 24 in. thick and 72 in. in dia. It is clad with a Ni-base alloy (AWS ERNiCr-3) wherever an attachment is welded on. All of the attachments are 304 series stainless steel except for an Inconel 600 attachment which forms a skirt around the bottom of the plug. The manufacturing sequence for the RRP called for the welding to be accomplished in three phases.

- 5411 BOUNDARY-VALUE THERMAL-HYDRAULIC SOLUTION IN A REACTOR FUEL BUNDLE. Sha, W.T.; Schmitt, R.C.; Huebotter, P.R. (Argonne Natl Lab, Ill). Am. Soc. Mech. Eng., [Pap.]; No. 75-HT-48, 13(1975).

A new computational model for steady-state, single-phase, thermal-hydraulic, multichannel analysis is presented. The model accounts for the conservation of mass, energy, and momentum subject to pressure-drop boundary conditions and leads to a nonlinear multipoint boundary-value problem. The turbulent interchange, radial thermal conduction and forced flow due to the wire-wrap or grid between the channels

are explicitly taken into account. The most important features of this new computational model are that: the effect of axial flow area variation has been incorporated into the derivation of governing equations; the crossflow approximation has been improved so that the assumption of constant transverse momentum flux in the direction under consideration is removed; and partial flow blockage occurring anywhere along the flow path can be analyzed, and the effect on the inlet mass velocity redistribution taken into account. 39 refs.

- 5412 THERMAL/HYDRAULIC PERFORMANCE TEST RESULTS OF SODIUM-HEATED STEAM GENERATOR MODEL EMPLOYING A DUPLEX TUBE. Waszink, R.P.; Hwang, J.Y.; Efferding, L.E. (Westinghouse Electr Corp, Tampa, Fla). Am. Soc. Mech. Eng., [Pap.]; No. 75-HT-65, 11(1975).

The paper presents thermal/hydraulic performance results of testing a single duplex tube sodium-heated steam generator model in the GPL-1/SWL-1 (sodium/water loops) test facility at the Westinghouse Advanced Reactors Division. The Small Steam Generator Model (SSGM) was designed, built and tested to proof-test a single tube of duplex wall construction at conditions identical to the alternate concept evaporator for the Clinch River Breeder Reactor Plant (CRBRP). 21 refs.

- 5413 PRELIMINARY THERMAL-HYDRAULIC DESIGN AND PREDICTED PERFORMANCE OF THE CLINCH RIVER BREEDER REACTOR CORE. Carelli, M.; Markley, R.A. (Westinghouse Adv React Div, Madison, Pa). Am. Soc. Mech. Eng., [Pap.]; No. 75-HT-71, 8(1975).

The preliminary thermofluids design of the core assemblies of the Clinch River Breeder Reactor Plant (CRBRP) is presented along with the predicted performance for various plant operating conditions. The selected radial blanket assemblies shuffling scheme and the new approach adopted for fuel and blanket assemblies orificing is discussed. Relevant thermofluids parameters (assembly flow rate and mixed mean outlet temperature, fuel rod temperatures, fission gas plenum pressure, etc.) are presented. Highlights of a detailed uncertainties study performed are summarized. This preliminary analysis has shown that the CRBRP core behavior is compatible with design limits and requirements; future developments for a final analysis are outlined. 4 refs.

- 5414 PREDICTION OF LOCAL AND INTEGRAL TURBULENT TRANSPORT PROPERTIES FOR LIQUID-METAL HEAT TRANSFER IN EQUILATERAL TRIANGULAR ROD ARRAYS. Ramm, H.; Johannsen, K. (Tech Univ Berlin, Ger). Am. Soc. Mech. Eng., [Pap.]; No. 75-HT-NN, 1-6(1975).

A theoretical method based on a phenomenological turbulence model has been applied to evaluate turbulent transport properties for liquid-metal heat transfer in bare equilateral triangular rod bundles. Results obtained for local distributions of thermal eddy diffusivity in the various directions are presented in terms of correlations. From a subsequent solution of the three-dimensional heat transfer problem between two characteristic interior subchannels under conditions characteristic for tracer-type mixing experiments, integral thermal mixing coefficients and thermal length scales have been evaluated. 10 refs.

- 5415 BRAZED THERMOCOUPLE PASS-THROUGH FOR SODIUM SERVICE IN A LIQUID-METAL-COOLED FAST

BREEDER REACTOR. Walker, D.E. Dearborn, MI; Society of Manufacturing Engineers (1975). 7p.

Sensors installed in special fuel elements for the EBR-II reactor have 30 ft long leads that pass from the Na environment through a sealed bulkhead. An H atmosphere, induction heated brazing furnace has been constructed to braze 20 to 26 separate sensor leads at one time. The brazed seals are leak-tight and the sheath wall has less than 10 percent interaction with the braze alloy.

5416 WE ALMOST LOST DETROIT. Fuller, J.G. New York; Reader's Digest Press (1975). 280p.

This is the documented, true account according to the author, of what happened on the afternoon of October 5, 1966, when the control panel inside the Enrico Fermi atomic reactor near Detroit, Michigan, suddenly registered high radiation levels, a sign of critical danger. Detroit was thirty miles away. The Fermi accident and the others described in this book demonstrate the fact that no matter how much diligence is exercised in the design, construction, and operation of reactors, things can go wrong. What is forgotten is that the public had a right to know these risks years ago, when the initial decisions regarding the acceptability of nuclear power were being made. Only now is it learned that the public was deliberately misled and deceived by the former Atomic Energy Commission regarding the possibility of major nuclear reactor accidents and the potential consequences of such accidents, states Mr. Carl J. Hovevar in the Foreword. The book surveys the accidents that have occurred in nuclear power stations here and abroad. (MCW)

5417 COMPATIBILITY OF A URANIUM METALLIC FUEL WITH OKH16N15M3B CHROME-NICKEL STEEL. Sergeyev, V.S.; Karpov, V.M.; Anuchkin, A.M. Sostoyaniye I Perspektivy Rabot Po Sozdaniyu Aes S Reaktorami Na Bystrykh Neytronakh; 2: 542-557(1975). (In Russian).

The designers of fast reactors are confronted with the question of the necessity of creating economically feasible fuel elements. One of the possible ways of solving this problem is the use of metallic uranium and uranium-plutonium fuel. They report that upon irradiation of uranium-plutonium fuel the achievable density of fission was equal, and in certain instances, exceeded the density of fission for oxide and carbide fuel without any notable change in the form of the samples. They investigated the mechanism of diffusion interaction of OKH16N15M3B steel with uranium and several alloys on its base for the purpose of seeking the possibility of ensuring compatibility of the fuel with the shell at temperatures up to 700--750°C. The samples were annealed in a temperature range of 400--800°C in a vacuum of $5 \cdot 10^{-5}$ mmHg. They give the microstructures of the zones of interaction between uranium and steel, obtained at various temperatures after 100 hours of diffusion annealing. Annealing at a temperature of 750°C and above leads to the formation of a eutectic and fusion of the samples. Preliminary investigations showed that the protective barriers of such materials as V, Zr, Nb, Ta, and Mo may ensure long-term compatibility at temperatures up to 700--800°C.

5418 PARAMETER OPTIMIZATION OF AES WITH A LARGE FAST REACTOR. Rineyskiy, A.A.; Orlov, V.V.; Subbotin, S.A.; Yarovitsin, V.V.; Novozhilov, A.I. Sostoyaniye I Perspektivy Rabot Po Sozdaniyu Aes S Reaktorami Na Bystrykh

Neytronakh; 1: 152-159(1975). (In Russian).

The temperature and pressure of the AES steam power cycle have a substantial influence on the physical characteristics of the LMFBR and on the economic indicators of AES as a whole. By means of the optimization program ROKBAR, a study was made of the influence of the temperatures of the heat-transfer agent at the output from the LMFBR on the doubling time (T_2) and reproduction coefficient (KV). A study was made of the range of preheatings of the heat-transfer agent (sodium) 150 to 240°C, which corresponds to output temperatures of 500 to 530°C. The thermal power for all variants was assumed as 4300 Mw. Plutonium from a thermal reactor was used as fuel, with the initial isotope composition: $^{239}\text{Pu}/^{240}\text{Pu}/^{241}\text{Pu} = 0.64/0.25/0.11$. The time of the external fuel-supply cycle was assumed as 0.5 years. Results are given of estimates of the effect of warming up the heat-transfer agent on the characteristics of a high-power LMFBR, particularly on the economic characteristics depending on the efficiency and preheating.

5419 HIGH POWER FAST REACTORS WITH COMBINED USE OF URANIUM-235 AND PLUTONIUM. Ilyunin, V.G.; Kuz'min, A.M.; Murogov, V.M.; Orlov, V.V.; Silayev, Y.V.; Troyanov, M.F.; Shmelev, A.N.; Yurova, L.N. Sostoyaniye I Perspektivy Rabot Po Sozdaniyu Aes S Reaktorami Na Bystrykh Neytronakh; 3: 20-36(1975). (In Russian).

The efficiency of combined use in an LMFBR of ^{235}U and ^{239}Pu was studied. The efficiency is related to the increase in unit capacities of the LMFBR up to 1000-2000 Mw, which in turn is expedient from the standpoint of reducing the specific capital investments and from the standpoint of the economics of the fuel-supply cycle. Combined use of U and Pu in the active zone of the LMFBR is one of the possible ways of ensuring stability and efficiency in flattening the profile of the heat release field; the value of the internal coefficient of reproduction of reactivity in this case is close to unity, since in the internal subzone a fuel (plutonium) is produced which is around 30 percent more efficient than the depleted ^{235}U ; a more negative value of the sodium coefficient of reactivity is ensured. This permits an increase in the length of operating the LMFBR without reloading and considerable rise in the power.

5420 MONITORING OF OXYGEN AND HYDROGEN IN A SODIUM CIRCULATION LOOP BY ELECTROMECHANICAL MEASURING METHODS. Kozlov, F.A.; Kuznetsov, E.K.; Kozub, P.S.; Ul'man, F.A.; Rets, T.; Teske, K. Sostoyaniye I Perspektivy Rabot Po Sozdaniyu Aes S Reaktorami Na Bystrykh Neytronakh; 2: 280-308(1975). (In Russian).

The article concerns measurement methods and instruments developed at the Rossendorf (GDR) Central Nuclear Research Institute. The instruments were installed and tested on a sodium loop in the FEI [R-and-D facility]. The authors verified the research on measuring cells in various exploitational conditions of the sodium circulation loop and compared the results obtained with the data from other methods of measuring impurities. They give the scheme and describe the experimental installation. They justify the choice of working parameters of the measuring cells for oxygen (temperature of 350 $\pm$ 5°C, consumption of sodium 0.2 m³/h, of air 2 to 4 l/h, rate of temperature increase of sodium 2°C/min), for which they conducted experiments on finding the influence of the consumption of sodium, consumption of air, and sodium temperature on the indications of the cells. They evaluate the stability of the indications of the cells and

the time of operation, the influence of hydrogen on the activity of oxygen in sodium. The measuring cell for oxygen and hydrogen operated in conjunction with a diffusion nickel membrane with a surface of 1750 cm² and thickness of 1 mm. The temperature of the membrane was 450°C, the consumption of the carrier gas (argon) was 6 l/h. They give examples of monitoring the working regime of the loop by electrochemical methods of measurement.

- 5421 DEVELOPMENT OF EXPERIMENTAL IRRADIATION DEVICES FOR MATERIALS TESTING IN THE BOR-60 REACTOR. Samsonov, B.V.; Shulimov, V.N.; Drutikova, V.V.; Seredkin, S.V.; Mel'der, R.R. Vopr. Atom. Nauki Tekhn. Ser: Radiatsion. Materialoved., Metokika Tekhn. Oblucheniya; 5: 7-13(1975). (In Russian).

The start-up of the industrial test BOR-60 fast-neutron reactor made it possible to greatly expand the breadth of research into radiation materials technology. It became possible to resolve such questions as finishing off fuel cells of newly-built fast-neutron reactors; study of radiation swelling of stainless steels, study of corrosive properties of materials in a flow of sodium under the influence of reactor irradiation, and some other questions. In order to perform the entire system of studies, experimental devices are necessary to permit irradiation of materials under different conditions. Methodology questions of design of irradiating devices for fast-neutron reactors are considered. The design of various types of materials technology packs, developed by the BOR-60 reactor, is described.

- 5422 PROPERTIES OF STABILIZED CHROME-MOLYBDENUM STEELS FOR STEAM GENERATORS OF FAST-NEUTRON REACTORS WITH A SODIUM HEAT CARRIER. Ilinchev, G.; Mayer, F.; Fresl, M. Sostoyaniye Perspekt. Rabot Sozlanuyu Aes Reaktor. Bystrykh Neytronakh; 2: 425-458(1975). (In Russian).

Mechanical properties and corrosion resistance were of steels on a base of 10Kh2 and 25Mn, stabilized with Nb and Ti in various combinations. Influence of oxygen on the transport of carbon was determined during 1000-h tests in liquid sodium containing about 2.10^{-4} or 3.45 to 6.58×10^{-2} mass percent oxygen. The corrosion resistance was studied at temperatures from 400 to 700°C (800°C for pure sodium). Furthermore, the structural stability, mechanical properties and impact viscosity at high resistance, weldability of the steels were studied and a microanalysis and phase analysis of the carbides were made. The tests showed that the investigated steels are distinguished by a low activity of carbon and therefore do not decarbonize under the effect of liquid sodium and at high temperatures (up to 600 to 700°C) also do not cause carbonization of the stainless chrome-nickel steels. The mechanical and technological properties of the stabilized steels were compared with the properties of the unstabilized steel type 10Kh2M and were satisfactory up to a temperature of 575°C. This makes it possible to use them as structural material for steam generators in high-speed reactors for the purpose of raising their working parameters and efficiency. 10 figures, 2 tables, 15 references.

- 5423 STUDY OF THE SIGNALS MEASURING THE TEMPERATURE OF SODIUM IN A NUCLEAR REACTOR. Joly, F.; de Lapparent, D.; Dubuisson, B.; Gaillard, P. (Univ de Technol de Compiègne, Fr). Automatisme; 20: No. 6/7, 268-

275(1975). (In French).

A method is proposed for studying signals measuring the temperature of sodium in the nuclear reactor Rapsodie. Each signal is viewed as a vectorial gaussian process, the properties of which are studied. The use of this signal makes it possible to assure monitoring of the thermocouples of the reactor core and discovery of abnormal phenomena.

- 5424 APPLICATION EXPERIENCE WITH CODES AND STANDARDS IN THE DESIGN OF HEAT EXCHANGERS FOR LIQUID METAL AND GAS COOLED REACTOR SYSTEMS. Pal, D.H.; O'Donoghue, J.K.; Stenner, J.R. (Foster Wheeler Energy Corp, Livingston, NJ). Am. Soc. Mech. Eng., [Pap.]; No. 75-Pwr-16, vp(1975).

This paper highlights some of the application experience with various codes and standards in the design of heat exchangers for liquid metal fast breeder reactor and high temperature gas cooled reactor systems. Application experience in the areas of elevated temperature design/analysis and classification of component parts and in-service inspection are discussed. Areas of deficiency in the current codes and standards are identified along with corrective recommendations.

- 5425 FLOW-INDUCED VIBRATIONS OF SPIRAL WIRE-WRAPPED FUEL ASSEMBLIES. Kinsel, W.C. (Westinghouse Hanford Co, Richland, Wash). Am. Soc. Mech. Eng., [Pap.]; No. 75-WA/HT-76, vp(1975).

The vibration characteristics of 217-pin spiral wire-wrapped fuel assemblies were investigated for a variety of flow rates, water temperatures and structural support conditions. The test results were used to optimize the mechanical design of fuel assemblies for the Fast Flux Test Facility (FFTF). The pin displacements were generally low (e.g., 5 mm or less), except for certain support conditions. Pin vibrations were characterized by distinct resonances at low flow rates; however, at high flow rates the response spectra were characteristic of broad-band Gaussian distributions. Both pin and shroud duct displacements were approximate functions of the square of the flow velocity. Water temperature had little effect on vibration levels. 11 refs.

- 5426 THERMAL/HYDRAULIC DESIGN CONSIDERATIONS FOR CLINCH RIVER BREEDER REACTOR PLANT INTERMEDIATE HEAT EXCHANGERS. Aburonia, M.M.; Cho, S.M.; Sawyer, R.H. (Foster Wheeler Energy Corp, Livingston, NJ). Am. Soc. Mech. Eng., [Pap.]; No. 75-WA/HT-101, vp(1975).

The heat transport system of the Clinch River Breeder Reactor Plant (CRBRP) includes intermediate heat exchangers (IHX's) to form barriers between the primary radioactive and secondary nonradioactive sodium coolant loops. The IHX is a vertical, cross/counter flow, shell-and-tube heat exchanger for high-temperature liquid-sodium service and its thermal rating is 325 Mw per unit. The secondary sodium flows vertically upward in 2850 parallel tubes and is heated by the downward flowing primary sodium on the shell side. This paper outlines the thermal and hydraulic design criteria and analyses for the IHX under all operating conditions; normal full and part load, startup and shutdown, upset and emergency events. 7 refs.

- 5427 LMFBR STEAM GENERATORS AREAS REQUIRING SPECIAL ATTENTION. McDonald, J.S. (Rockwell Int, Canoga Park, Calif). Am. Soc. Mech. Eng., [Pap.]; No. 75-WA/HT-104, vp(1975).

The steam generator is of vital importance to the economics and reliability of Liquid Metal Fast Breeder Reactor power plants. The properties of sodium, as well as the adverse effects which can result from tube leaks and associated Na-H₂O reactions, introduce a number of special considerations in the design of sodium heated steam generators. These special concerns are discussed in this paper, and the major problems which have occurred in liquid metal heated steam generators are reviewed. 5 refs.

- 5428 APPLICATION OF THE ASME BOILER AND PRESSURE VESSEL CODE AND RDT STANDARDS TO THE CLINCH RIVER BREEDER REACTOR PLANT SODIUM-HEATED STEAM GENERATOR. Parkman, M.F.; Watson, F.H.; Whipple, J.C. (GE, Sunnyvale, Calif.). Am. Soc. Mech. Eng., [Pap.]; No. 75-Pwr-19, vp(1975).

The Clinch River Breeder Reactor Plant sodium-heated steam generators are in the early stages of design and material procurement to the requirements of Section III of the ASME Code (Winter 1974 Addenda), elevated temperature Code Cases, and supplementary RDT Standards. Major features of the steam generator are described relative to application of the Code and Standards. Extensive modifications to RDT Standards have occurred, and new, simplified RDT M Standards for material have been issued. Problems in coordinating the Code and RDT Standards with the steam generator. Design Specification and long lead time materials procurement are outlined.

- 5429 FFTF OUTLET NOZZLE DESIGN CONSIDERING NATURAL AND FORCED CONVECTION FLOW IN PARALLEL ANNULI, WITH HEATING. Kinyon, B.W.; Vroom, D.W. (Combust Eng Inc, Windsor, Conn.). Am. Soc. Mech. Eng., [Pap.]; No. 75-WA/HT-105, vp(1975).

The FFTF reactor vessel is protected from the 1050 F (565.6 C) sodium in the outlet plenum by providing a bypass annulus fed by the 700 F (371.1 C) inlet sodium. The hot sodium is ducted across the bypass annulus into the outlet nozzles, which must have a smooth axial temperature gradient between the hot outlet pipe and the cooler vessel wall. To achieve the proper temperature profile, there is a liner between the nozzle and the outlet sodium duct, forming two annuli for out-flow of bypass sodium into the outlet pipe. Restricting orifices in each annulus control this bleed flow. A computer program designed to test the validity of the assumption of pure axial flow considers circumferential flow due to density differences as well as the axial flow due to the axial pressure gradients, and includes the heat transferred from the outlet sodium to the bleed sodium in the nozzle annuli. 2 refs.

- 5430 U.S. REACTOR DEVELOPMENT PROGRAM. Kintner, E.E. (Energy Research and Development Administration, Washington, DC). pp 3B.3.1-3B.3.4 of Proceedings of 1975 IEEE southeastern Region 3 conference. Volume I. Electricity: an expanding technology. New York; Inst. of Electrical and Electronics Engineers, Inc. (1975).
From IEEE southeastern; Charlotte, NC, USA (7 Apr 1975).

To meet a rapidly increasing demand for energy, the U.S. strategy is directed toward making the best use of all available resources. Nuclear power has a key role in this strategy and will be required to supply an ever larger share of electricity. Advanced reactors, or breeders, will be required to assure a long-

term energy supply. Consistent with this requirement, development of the Liquid Metal Fast Breeder Reactor (LMFBR) has been given priority in the AEC's broad reactor development program.

- 5431 MONITORING OF IMPURITIES IN LIQUID METAL FAST BREEDER REACTORS (LMFBR). Roy, P. (General Electric Co., San Jose, CA). pp 919-920 of Metal-slag-gas reactions and processes. Foroulis, Z.A.; Smeltzer, W.W. (eds.). Princeton, NJ; The Electrochemical Society, Inc. (1975).

From 147. symposium on metal-slag-gas reactions and processes; Toronto, Canada (11 May 1975).

The present state-of-the-art of oxygen and carbon monitoring devices to be used in liquid sodium is presented. (FS)

- 5432 ELECTROCHEMICAL OXYGEN METERS FOR LIQUID SODIUM FAST BREEDER REACTOR SYSTEMS. Berkey, E.; Reed, W.H. (Westinghouse Research Labs., Pittsburgh, PA). pp 921-922 of Metal-slag-gas reactions and processes. Foroulis, Z.A.; Smeltzer, W.W. (eds.). Princeton, NJ; The Electrochemical Society, Inc. (1975).

From 147. symposium on metal-slag-gas reactions and processes; Toronto, Canada (11 May 1975).

An oxygen meter was developed for use in the sodium coolant system. The meter consists of a high purity closed-end Y₂O₃ doped ThO₂ electrolyte ceramic tube containing air. The electrolyte tube is placed directly in the Na stream. It functions as a reliable, on-line measuring device. (FS)

- 5433 POLICY REPORT ON HIGH LEVEL RADIOACTIVE WASTES. Santa Fe, NM; Energy Resources Administration (1975). 81p. (NP-23948). Executive Office of the Governor, Energy Resources Administration, Santa Fe, New Mexico.
Report from the Governor's Task Force Committee.

For the near term, ERDA and the NRC believe bedded salt formations offer the best approach for implementing high-level radioactive waste disposal. Salt formations will likely be used as the repository site, with those in Southeastern New Mexico the most likely to be selected. If that happens, New Mexico must address the social issues that are bound to arise. Any ultimate waste disposal scheme that consumes regional resources such as land must pay the opportunity cost of consumption in perpetuity. A federal waste repository, of itself, will provide very little benefit to a region. Little local employment will result and other economic impacts will be minimal. There is no compelling reason for the people of New Mexico to accept a high-level waste repository. The recommendations are made: that a committee of technical competent individuals should be appointed by the Governor and the Energy Resources Board to oversee this federal work in the southeastern salt beds, and that New Mexico should do its own risk-benefit analysis of the high-level radioactive waste repository.

- 5434 CESIUM VAPOR CYCLE FOR AN ADVANCED LMFBR. Fraas, A.P. (Oak Ridge Natl Lab, Tenn.). Am. Soc. Mech. Eng., [Pap.]; No. 75-WA/Ener-5, 1-12(1975).

A cesium vapor topping cycle offers a promising means for coupling an advanced liquid metal-cooled fast breeder reactor to a steam system. If compared with a conventional sodium intermediate circuit, it yields an increase in thermal efficiency of about five points,

possesses important advantages from the maintenance standpoint, and greatly reduces the likelihood of serious troubles that might arise if a leak were to develop in the steam generator.

- 5435 TESTING AND EVALUATION OF ELASTOMERIC SEALS FOR SODIUM-COOLED FAST REACTOR SYSTEMS. Welch, F.H.; Kurzeka, W.J. (Rockwell Inst, Canoga Park, Calif). Am. Soc. Mech. Eng., [Pap.]; No. 75-WA/NE-1, 1-8(1975).

The Fast Flux Test Facility and the Clinch River Breeder Reactor Plant require elastomer seals for safe, reliable operation; these seals, with minimal maintenance, must be capable of preventing the leakage of radioactive gases to the atmosphere. A program was established to select commercial elastomers in standard shapes, such as O-rings, T-seals, Quad-rings and Q-seals, and to determine their pertinent characteristics. Buna N, ethylene propylene, polyurethane, and silicone elastomers were tested for their permeability, leakage, and compression set. The test variables included time, temperature, surface finish, percent compression, gas type, and lubricant. A method was developed for predicting gas leakage for permeability data.

- 5436 TEN YEARS' MAINTENANCE EXPERIENCE AT EBR-II. Buschman, H.W.; Hutter, E.; Leman, J.D.; Moriarty, K.J. (Argonne Natl Lab, Idaho Falls, Idaho). Am. Soc. Mech. Eng., [Pap.]; No. 75-WA/NE-3, 1-8(1975).

EBR-II has operated for over 10 years. This period has provided considerable experience in the maintenance and reliability of components and systems in a Liquid Metal Fast Breeder Reactor. This experience has provided a basis for extending the operation of EBR-II for another 10 years.

- 5437 ASSESSMENT OF ECONOMIC INCENTIVES FOR THE LIQUID METAL FAST BREEDER REACTOR. Stauffer, T.R.; Wyckoff, H.L.; Palmer, R.S. (Harvard Univ, Cambridge, Mass). Am. Soc. Mech. Eng., [Pap.]; No. 75-WA/NE-5, 1-9(1975).

This paper assesses the economic rationale for continuing the national commitment to the development of a fast breeder reactor. The question, "to breed or not to breed," is multifaceted, involving complex and conflicting considerations of cost, safety, and environmental impact for each competing electrical generation strategy. The discussion is focused upon estimating the savings in electricity costs which might ensue from the successful development of a commercial fast breeder reactor.

- 5438 LMFBR HARDWARE AND COMMERCIAL STANDARDS. Rowan, W.J.; Winkleblack, B.K. (Westinghouse Electr Co, Madison, Pa). Am. Soc. Mech. Eng., [Pap.]; No. 75-WA/NE-6, 1-4(1975).

This paper states the goals of the development program for Liquid Metal Fast Breeder Reactors (LMFBR's) and outlines the design, engineering, and documentation approaches being used. There is significant design and development work to be accomplished on the Clinch River Breeder Reactor (CRBRP); however, no technology breakthroughs are required since much of the component work consists of scaling up proven designs. Careful documentation is laying the groundwork for a commercial liquid metal fast breeder industry.

FACILITY. Toepel, F.P.; Moody, J.C. (R Ind, Energy Prod Div, Santa Ana, Calif). Am. Soc. Mech. Eng., [Pap.]; No. 75-WA/NE-2, 1-15(1975).

This paper covers selected features of a roller nut type Control Rod Drive Mechanism (CRDM) which was designed for the Fast Flux Test Facility (FFTF). A detailed discussion is given of each of the principal components which underwent special design and development to meet the unique environment and requirements of the sodium cooled Fast Test Reactor. The CRDM was developed by Royal Industries under contract to Westinghouse Electric Co., Advanced Reactors Division.

- 5440 SAFETY AND ENVIRONMENTAL IMPACT OF LIQUID METAL FAST BREEDER REACTORS. Buhl, A.R.; Graham, J. (Energy Res and Dev Agency, Knoxville, Tenn). Am. Soc. Mech. Eng., [Pap.]; No. 75-WA/NE-7, 1-4(1975).

Liquid metal fast breeder reactors are a necessary component in the national energy program planned for introduction as commercial power generating stations in the mid 1980's. This paper treats the safety and environmental impact of liquid metal fast breeder reactors by considering principally the matters of concern, most of them expressed by critics of the program in areas of safety and environmental impact. Areas of major concern, such as the use of sodium and its potential loss from the system as a coolant and the potential for critical reconfiguration accidents involving an energy release or damage to the core are considered. Other areas of concern such as plutonium toxicity and the potential for inadvertent or malicious release of plutonium, waste management and waste heat rejection, and accidental releases of radioactivity are also covered.

- 5441 RISK AVERSION AND ENERGY POLICY: A CASE FOR BREEDER RESEARCH AND DEVELOPMENT. Hamester, H.L.; Graves, G.A.; Plummer, J.L. (Natl Sci Found, Washington, DC). Energy Syst. Policy; 1: No. 3, 233-258(1975).

The continuance of breeder research and development can be argued at many levels. One level of argument views breeder RandD as an "insurance policy" to satisfy society's aversion to the following two categories of risk: (1) that without a commercial breeder capability by the 1990s, the electric power industry might be intolerably vulnerable to scarcities in environmentally acceptable uranium ore; and (2) that no other advanced central station electric power technology besides the breeder presently shows sufficient promise of being available for unsubsidized commercialization by the 1990s. This paper primarily addresses the first of these two categories of risk. The specific topics covered are electricity demand growth, uranium supply, advanced convertors versus breeders, program financing and management, and the relationship of benefit-cost analysis to breeder decision-making. The point is made that proceeding with breeder RandD does not commit the nation to breeder commercialization.

- 5442 HEAT CONDUCTIVITY BETWEEN UO₂SUB 2% CORE AND CASING IN FAST REACTOR FUEL ELEMENTS. Lebedev, I.G.; Golovnin, I.S.; Bagdasarov, Yu.E.; Kuz'min, V.I.; Bibilashvili, Yu.K.; Tikhomirov, B.B.; Kalashnik, G.V.; Silakov, A.S. Sov. At. Energy (Engl. Transl.); 37: No. 6, 1227-1231(Dec 1974).

The results of investigations of the heat conductivity coefficient between UO₂sub 2% core and OKh16N15M3B steel casing and a temperature

- 5439 DEVELOPMENT OF A ROLLER NUT CONTROL ROD DRIVE MECHANISM FOR THE FAST FLUX TEST

of $\pm$ similar ± 500 C at the inner surface of the casing are presented. 7 refs.

- 5443 HOW BOR-60 REACTOR POWER STATION EQUIPMENT IS SERVICED IN A RADIATION ENVIRONMENT. Kizin, V.D.; Polyakov, V.I.; Chechetkin, Yu.V.; Kevrolev, V.P.; Levin, L.M.; Rybakov, V.N. Sov. At. Energy (Engl. Transl.); 37: No. 6, 1246-1250 (Dec 1974).
BOR-60 REACTOR; REACTOR MAINTENANCE

- 5444 OPTIMIZATION OF THE DOUBLING TIME T_{SUB} 2% AND THE FUEL COMPONENT OF CALCULATED EXPENDITURES E_{SUB} P_{S} IN FAST REACTORS BY MEANS OF PARTIAL RECHARGING. Grachev, E.A.; Deinega, N.L. Sov. At. Energy (Engl. Transl.); 37: No. 3, 914-919 (Sep 1974).
5 refs.

- 5445 DEVICE FOR TUNING THE ACOUSTIC CONTROL SYSTEM TO DETECT LIQUID-METAL COOLANT BOILING. Mitenkov, F.M.; Pakhomov, V.V.; Merkulov, V.I.; Artem'ev, L.N.; Karasev, V.N.; Kulikov, V.V. USSR Patent 441,596/A/. 30 Aug 1974. 3p. (In Russian).
A device for tuning a system of acoustic control of boiling of liquid metal coolant in a fast reactor is described. The device comprises a suspension with a hollow current-conducting bus installed thereon, and a casing with a heat element. In order to improve tuning accuracy, units for measuring the voltage drop in respective zones are installed along the entire heat element. The bus is connected to the middle section of the heater. The device may be placed in one of the core channels via the reactor cover and allows to obtain controlled boiling where it is anticipated during a local breakdown and in the presence of basic noises produced by the device.

- 5446 PIPING FOR NUCLEAR POWER REACTORS. REVIEW. Rodchenkov, B.S. At. Tekh. Rubezhom; No. 8, 3-10 (Aug 1974). (In Russian).
The article reviews production technology, inspection methods, and general characteristics of pipes and tubing: 1 - for piping of the primary loop made of austenitic stainless steels type 304 and 316, and of piping running from the reactor to the steam generator or directly to the turbine in which case it is made of bimetallic materials for PWR and BWR reactors; 2 - for the cladding of fuel elements of PWR and BWR reactors which is made of zirconium alloys. Thus the technology of the Japanese forms "Kobe Steel" and "Sumimoto Metal Industry" is described as it is applied to the fabrication of fuel element cladding for the "Fukushima-2" and "Mihama-2" reactors. This has produced piping of higher corrosion resistance, of more favorable hydrides orientation and more stringent allowances in fabrication; 3 - for fuel element cladding of the magnox type for graphite reactors cooled with CO_2 , for gas cooled reactors with elevated coolant temperatures (over 500°) made of chromenickel steel stabilized with titanium or niobium; 4 - for the fuel element cladding of fast reactors that are cooled with liquid sodium, where the cladding is made of type 316 stainless steel. The particularities of fabrication of channel tubing made principally of zircaloy-2 and zirconium-niobium alloy are given as well as the quality control required for the metal. Data are given on the types of materials used, on the dimensions and pressures used in the channel tubing made of zirconium alloys.

- 5447 ACOUSTIC TRANSDUCER AND ITS PROCEDURE OF FABRICATION. Smith, R.W.; Adamson, G.E. (to USAEC Directorate of Licensing, Washington, D.C.). French Patent 2,296,986/A/. 26 Jul 1974. Priority date 27 Jul 1973, United States of America (USA), 12p. (In French).
The invention relates to an acoustic transducer able to endure the high temperatures, high levels of radiations and the surrounding caustic media involved inside fast breeder reactors, said transducer being intended for reliable sodium control. A new process insures a continuous maximum acoustic coupling at the interface between the active piezoelectric element (made of ceramics, lead zirconate or lead titanate) and the face of the transducer. Said interface is obtained by coating the juxtaposed facing plate of the transducer and active surface of the element with wetting agents compatible with the bonding alloy subsequently used in view of ensuring the acoustic bonding.

- 5448 NUCLEAR POWER PLANT. Grinman, M.I.; Kruglikov, P.A.; Povolotskii, O.B.; Smolkin, Yu.V.; Rineiskii, A.A. USSR Patent 431559/A/. 5 Jun 1974. 3p. (In Russian).
A method is proposed for increasing the efficiency of vapour-moderated intermediary superheating in a power plant with a fast sodium-cooled reactor. In this case, the intermediary superheater on the side of the heating vapour is connected to a steam line located between the evaporator and the steam superheating sections of the steam generator. The line for draining heated steam from the intermediary superheater is connected to a regenerative heater. This arrangement provides better efficiency of intermediary superheating and promotes cheaper operation.

- 5449 FAST NUCLEAR REACTOR. Zhan-Marl Berniol; Sabino Gallo. USSR Patent 597,351/A/. 6 Mar 1974. 5p. (In Russian).
The invention may be used to improve the design of a pool type liquid metal cooled fast reactor. There is a vertical main tank containing components of the primary cooling circuit, and an internal tank enclosing the core. The invention is aimed at increasing the stability of liquid metal thermohydraulic flow inside the main tank of power reactors (1000 MWe and more) that must improve the reliability. The aim is achieved by attaching splitters to the main tank wall and locating the upper part of the lateral wall of the internal tank below the top of the core and above the level of the heat exchanger outlets. In this case the external side of the lateral wall is bent downward and attached directly to the main tank.

- 5450 STEADY TEMPERATURE DISTRIBUTIONS IN A MULTI-REGION NUCLEAR FUEL ROD. Krudener, R.M. (Therm Search Associates, Jackson, WY); Wolf, H. pp 239-262 of Tenth southeastern seminar on thermal sciences. Watts, R.G.; Sogin, H.H. (eds.); New Orleans, LA; Tulane Univ. (1974).

From 10. seminar on thermal sciences; New Orleans, LA, USA (11 Apr 1974).

Steady temperature distributions are presented for a multi-region nuclear fuel rod having a high-conductivity auxiliary path to conduct energy out to the cladding and thus to the coolant. The fuel material modeled was an 80-20 mixture of UO_2 and PuO_2 , respectively. A Reference Design power level of 18 kw/ft was selected and two sizes of center voids in the fuel pellet were investigated. It was shown that power levels of 27 kw/ft could be reached

with the maximum fuel temperature (4,880°F) still below the melt temperature (5,100°F) for a sodium coolant temperature of 1,000°F. The results were obtained by numerical solution of the elliptic conduction equation and are presented for representative values of the gap conductances between the fuel, auxiliary path, and cladding materials. Details of the numerical procedure are not given but are briefly described.

- 5451 INELASTIC ANALYSIS OF PIPELINES IN FFTF CLS MODULE. Pan, Y.S.; Jetter, R.I. (Rockwell International Corp., Canoga Park, CA). pp 235-251 of Pressure vessels and piping: analysis and computers. Tuba, I.S.; Selby, R.A.; Wright, W.B. (eds.). New York, NY; American Society of Mechanical Engineers (1974).

From Pressure vessels and piping-materials-nuclear conference; Miami Beach, FL, USA (24 Jun 1974).

The inelastic analysis of pipelines in the FFTF Closed Loop System is described. The analytical tool is the general purposed program, MARC-CDC. Specially developed pipe bend elements and the straight pipe elements are used to model the pipelines. The pipe bend element in the MARC system has the ability to predict the two-dimensional stress distribution in elbows of the pipeline. This ability permits a reasonably rigorous solution to the nonlinear behavior of pipelines. The analysis is performed for two representative lines in the system. Results of these analyses are presented and discussed. 15 references.

- 5452 SUPERHEAT IN FLOWING SODIUM. PAPER 3.10. Hammett, F.G. (Michigan Univ., Ann Arbor (USA). Dept. of Mechanical Engineering). pp 258-262 of Finite-amplitude wave effects in fluids. Bjoernoe, L. (Technical Univ. Denmark) (ed.). Guildford, England; IPC Science and Technology Press (1974).

From Symposium on finite-amplitude wave effects in fluids; Copenhagen, Denmark (20 Aug 1973).

State-of-the-art concerning bubble nucleation in molten sodium, particularly as it applies to sodium-cooled fast-breeder reactors, is discussed. In particular, the possibilities of significant 'superheat' in flowing sodium systems are discussed, along with the measurement of gas nuclei content in sodium and in water. It is believed that the development of this ability is requisite for further progress in this area.

- 5453 NUCLEAR REACTOR. Mishel, O.; Anri-Zhak, G.; Filipp, V. USSR Patent 499,845/A/. 7 Aug 1973. Priority date 8 Aug 1972, France, 4p. (In Russian).

A liquid-metal reactor cooling system is proposed, featuring a number of advantages over the prior designs. Along with insulation of the hot sodium inside the primary tank, a reduction in the volume of the hot sodium in the reactor is achieved, or an increase in the height of heat exchangers for a given volume. Another advantage offered by the invention resides in the possibility of insulating one or several heat exchangers. The arrangement is simple and convenient.

- 5454 (PNL-TR--356) FAST REACTOR PROGRAM REPROCESSING PROJECT-PYROCHEMICAL HEAD-END PRELIMINARY TECHNICAL AND ECONOMIC EVALUATION. Heylen, P.P. (Centre d'Etude de l'Energie Nucleaire, Mol (Belgium)). Mar 1973. Translation of BLG--478, ENI--67, BN--71238/060/

N/010 77p. Dep. NTIS, PC A05/MF A01.

This study has demonstrated the technical, economic and ecological possibilities of a pyrochemical head-end based on application of molten salts to the conditioning, transport and disaggregation of fuels irradiated in fast sodium-cooled reactors and reprocessed in a plant having a capacity of 1 t/day. Preliminary economic evaluation of a scheme with integrated use molten salts resulted in a total investment cost of 1692 million FB and a head-end reprocessing price (not including waste transport and conditioning) of 1318 FB/kg. However, some of the unit operations in this scheme are ready for immediate application.

- 5455 APPARATUS FOR CONTROLLING MOLTEN CORE DEBRIS. Golden, M.P.; Tilbrook, R.W.; Heylman, N.F. (to Energy Research and Development Administration). US Patent 4,036,688. 19 Jul 1972. Filed date 9 Apr 1975. 16p.

PAT-APPL-566,489.

Disclosed is an apparatus for containing, cooling, diluting, dispersing and maintaining subcritical the molten core debris assumed to melt through the bottom of a nuclear reactor pressure vessel in the unlikely event of a core meltdown. The apparatus is basically a sacrificial bed system which includes an inverted conical funnel, a core debris receptacle including a spherical dome, a spherically layered bed of primarily magnesia bricks, a cooling system of zig-zag piping in graphite blocks about and below the bed and a cylindrical liner surrounding the graphite blocks including a steel shell surrounded by firebrick. Tantalum absorber rods are used in the receptacle and bed. 9 claims, 22 figures.

- 5456 DEVICE FOR FAST REACTOR CHARGING. Rakhimov, V.V.; Artem'ev, L.N. USSR Patent 440,967/A/. 16 May 1972. 2p. (In Russian).

The invention refers to servomechanisms of industrial nuclear reactors with liquid metal coolant. In order to provide control over hermeticity of fuel elements assemblies during their transportation the wall of the directing tube of the transporting mechanism contains channels supplying gas for bubbling the alkali metal (coolant) with the outlet at the level of the fuel packet lower edge, and a channel for removing the gas, placed at sampling outlet.

- 5457 TRANSIT DEVICE FOR REACTOR FUELING. Saijo, T.; Hayashi, H.; Yamaki, H.; Sasaki, M. (to Hitachi Ltd., Tokyo (Japan)). Japanese Patent 1976-110195/A/. 10 Dec 1971. 3p. (In Japanese).

The object of the invention is to effect refueling without detracting from the safety of the nuclear container by providing a relay container which has two openings for communication with the inside and outside of the container and is also provided with cooling and heating means, a chain being provided with a bucket disposed for revolution between the inlet and outlet. A relay container filled with an inert gas and a nuclear container are connected together with a first fuel inlet and outlet passage provided with a door valve, and the relay container and outside of the nuclear container are connected together with a second fuel inlet and outlet passage having a door valve. Further, a transfer means including a fuel container capable of movement between both the passages is provided.

- 5458 DEVICE FOR HOLDING IN POSITION THE PINS

OF A CLUSTER INSIDE THE CASING OF AN ASSEMBLY INTENDED FOR NUCLEAR REACTORS. Boyer, P.; Lorient, P. (to CEA, 75 - Paris (France)). French Patent 2,301,070/A/. [nd]. 11p. (In French).

Said invention is intended for fissile or fertile fuel assemblies in fast reactors. It is related to an improvement to the devices supporting and holding in position the pins of a cluster inside an assembly, said devices ensuring the pins to be simultaneously locked in the vertical and transverse directions.

- 5459 (EURFNR--1412) INVESTIGATION OF THE THREE-DIMENSIONAL THERMOELASTIC DEFORMATION OF THE CORE ASSEMBLY OF A FAST BREEDER REACTOR UNDER STEADY-STATE OPERATING CONDITIONS. Hoang, Y.S. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Reaktorentwicklung). [nd]. Translation of KFK--1874 196p. Dep. NTIS, PC A09/MF A01.

Work performed under United States--Euratom Fast Reactor Exchange Program. Thesis. Submitted to Karlsruhe Univ., Ger..

A method is described which has been developed in order to calculate three-dimensional deformation of the reactor core, taking into account thermal expansion. The computation method developed is applied to the MARK I core of SNR 300. Core deformations are calculated under different support conditions for the subassemblies in the grid plate and in the upper clamping plane. The results show that the tangential components of the force and deformations are not always smaller than the respective radial components: the ratio of these two variables strongly depends on the support conditions selected. If the support loads of a radial row of subassemblies are compared with those of subassemblies lying on the same radius and peripheral row, clear difference can usually be seen.

- 5460 Rapports de la Commission Consultative pour la Production d'Electricite d'origine Nucleaire 1970-1974. (REPORTS FROM THE ADVISORY COMMITTEE FOR NUCLEAR POWER GENERATION, 1970-1974. VOL. II). Paris; Documentation Francaise ([nd]). 226p. (In French).

The present report is first dealing with the general status of nuclear power production throughout the world and is defining more accurately the French position. It is then dealing with the various aspects of the quantitative programme based on light water reactor types for the VIth plan, its justifications, the problems involved and the importance of this programme. The various items of a subsidiary programme are presented for the reactor types, the development of which is to be examined for the future (sodium cooled fast neutron reactors, heavy water reactors involving the EL600 project, and HTRs). Two fields closely related to nuclear power realizations are discussed: the fuel cycle and the effort required in research and development

- 5461 (DOE-tr--71) APPLICATION OF INTEGRAL TRANSPORT THEORY WITH LINEARLY ANISOTROPIC SCATTERING TO THE CALCULATION OF THE NEUTRON DISTRIBUTION IN FINITE PLATE ARRAYS OF FAST REACTORS. Boehme, R. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). [nd]. Translation of KFK--2501 133p. Dep. NTIS (US Sales Only), PC A07/MF A01.

Work performed under United States--W. Germany Fast Reactor Exchange Program. Thesis. Submitted to Univ., Karlsruhe,

Germany.

A method was derived to calculate from integral transport theory the neutron flux density close to boundaries between plate lattice regions of fast zero power reactors. Leakage parallel to the plate planes is treated by a momentum method. Linearly anisotropic scattering and the space dependence of effective cross-sections in the resonance region are taken into account by suitable approximations. The method was applied to evaluate reaction rate measurements in the vicinity of the core-blanket boundary of a SNEAK assembly which reproduced parts of the SNR 300 reactor. Using the cross-section set KFKINR agreement with the experiment was achieved for the space dependence of the ^{239}Pu fission rate and the ^{238}U capture rate.

- 5462 (DOE-tr--113) KADIS: A PROGRAM TO ANALYZE THE DISASSEMBLY PHASE OF HYPOTHETICAL ACCIDENTS IN (LMFBRs). Schmuck, P.; Jacobs, G.; Arnecke, G. (Kernforschungszentrum Karlsruhe G.m.b.H. (Germany, F.R.). Inst. fuer Neutronenphysik und Reaktortechnik). [nd]. Translation of KFK--2497, November 1977 207p. Dep. NTIS (US Sales Only), PC A10/MF A01.

Work performed under United States--West Germany Fast Reactor Exchange Program.

The KADIS program models the disassembly phase during power excursions in hypothetical LMFBR accidents. KADIS is based on point kinetics in the neutronics part and on a 2-dimensional representation of the reactor core in the hydrodynamics part. The core is modeled as an ideal, compressible fluid which is heated up adiabatically during the excursion. KADIS was built up with the help of the VENUS program of Argonne National Laboratory. Several important features were added to the basic VENUS model. A complete description of the mathematical models used is presented. The necessary information to handle the input/output of KADIS is included.

- 5463 (TNO-LMEC--29) 50 MW-TRIAL CIRCUIT. QUARTERLY REPORT, FIRST QUARTER 1978. (Nijverheidsorganisatie TNO, Apeldoorn-Zuid (Netherlands). Centraal Technisch Inst. TNO). [nd]. Translation of Kwartaalverslag 50 MW-beproevingscircuit 1e kwartaal 1978, April 1978 7p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States--TNO Technical Exchange Program.

During the entire period just completed the 50 MW installation has been in operation. Operations were specifically directed towards the testing program of the 85 MW-ZWT; primarily with the Na system and incidentally with the steam feedwater system.

- 5464 (TNO-LMEC--30) OPERATION 50 MW CIRCUIT. QUARTERLY REPORT, SECOND QUARTER 1978. (Nijverheidsorganisatie TNO, Apeldoorn (Netherlands)). [nd]. Translation of Kwartaalverslag 2e kwartaal 1978: exploitatie 50 MW-circuit, July 1978 10p. Dep. NTIS (US Sales Only), PC A02/MF A01.

Work performed under United States--TNO Technical Exchange Program.

The operational shutdown which had already started in the previous reporting period continued during this period. Full continuous service was resumed on March 13 and a start has been made on a program of pre-operational tests anticipating the completion of putting the system into service including the 85 MW-ZWT.

- 5465 PROCESS FOR STABILIZING DIMENSIONS OF DUPLEX STAINLESS STEELS FOR SERVICE AT ELEVATED

TEMPERATURES. US Patent Application 055,210.
[nd]. 11p.

Duplex stainless steel materials containing austenite plus delta ferrite, are dimensionally stabilized by heating the material to a reaction temperature between about 1050 to 1450°F (566 to 788°C), holding it at this temperature during transformation of delta ferrite to austenite plus sigma phase, and

subsequently heating to a reversion temperature between about 1625 to 1750°F (885 to 954°C), whereby the sigma phase transforms back to ferrite, but the austenite remains dispersed in the ferrite phase. Final controlled cooling permits transformation of ferrite to austenite plus sigma and, later, precipitation of carbides.